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Residual Radionuclide Contamination Within and Around Commercial Nuclear Power Plants

Assessment of Origin, Distribution, Inventory and Decommissioning

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Operated by
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U.S. Nuclear Regulatory
Commission

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ABSTRACT

The residual radionuclide concentrations, distributions and inventories at seven nuclear power plants (four shutdown and three operating) have been investigated to provide a data base for use in formulating policies, strategies and guidelines for the eventual decommissioning of retired nuclear power plants. This study has addressed radionuclides (both activation and fission products) transported from the reactor pressure vessel and deposited in all other contaminated systems of each nuclear plant. A companion study (NUREG/CR-3474) has addressed the neutron activated pressure vessel and its internals, plus the neutron activated concrete bioshield.

Emphasis has been placed on measuring the long-lived radionuclides which are of special concern from a low-level waste management standpoint, including ^{60}Co, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{94}Nb, ^{99}Tc, ^{129}I, ^{137}Cs, and alpha-emitting transuranic radionuclides with half-lives greater than five years. The measurement of these radionuclides is specifically required in 10CFR61, "Licensing Requirements for Land Disposal of Radioactive Wastes," to establish waste classification and subsequent disposal options.

The results of this study indicate that the most abundant long-lived radionuclides associated with contaminated piping, hardware and concrete for a period of up to about 10 to 20 years after shutdown generally included ^{60}Co, ^{55}Fe, ^{63}Ni, and ^{137}Cs. Large variations in the relative concentrations of these radionuclides existed from plant to plant. Their relative abundances were controlled by several factors, e.g. the major and trace element composition of primary loop construction materials, the primary coolant chemistry, the fuel integrity, and the corrosion product deposition processes. Cobalt-58, ^{54}Mn, and ^{65}Zn were frequently abundant short-lived radionuclides at the time of shutdown. Occasionally, ^{106}Ru, $^{141-144}$Ce and ^{110m}Ag were also present. Contamination residues normally contained very low concentrations of ^{90}Sr, ^{94}Nb, and isotopes of Pu, Am, and Cm. Iodine-129 and ^{99}Tc, being very soluble long-lived radionuclides, were generally not associated with residual radionuclide contamination to any significant degree. Cobalt-60 and ^{137}Cs will be the main contributors to the external whole body dose for several decades following shutdown. Residual radionuclide concentrations in the various plant systems decreased in the following order: 1) primary coolant loop, 2) rad-waste handling system, and 3) secondary coolant loop in PWR's and condensate systems in BWR's.

Radionuclide contamination of concrete in nuclear power plants is of two types: 1) surface contamination resulting from spills of radioactive materials and 2) neutron-activated concrete in the bioshield and floor directly underneath the pressure vessel. Surface contamination of concrete is extremely patchy and generally limited to areas of the plant where radioactive liquids have spilled. Cesium-137, ^{134}Cs, and ^{60}Co are the most abundant radionuclides in surface contaminated concrete. Cesium-137 and ^{134}Cs are preferentially sorbed onto bare concrete

relative to other radionuclides due to the ability of cesium to ion exchange with mineral phases in the concrete. This behavior was mainly noted for bare concrete surfaces or surfaces which had lost their paint coatings. Cesium-137 concentrations up to 3 μCi/gm have been observed in some of the most contaminated concrete. The decrease in radionuclide concentrations with depth in the concrete was dramatic and generally amounted to about 3 orders of magnitude over a depth of only 1 cm. Radionuclide concentrations in the neutron activated concrete directly below the pressure vessel extend several tens of centimeters deep, but at low concentrations, e.g. about 1100 pCi/gm of ^{60}Co, 2600 pCi/gm of ^{152}Eu, and 12,000 pCi/gm of ^{55}Fe at a depth of 8 cm in the concrete below the Turkey Point Unit 4 pressure vessel.

A significant observation is that essentially all of the contaminated piping and hardware (excluding the pressure vessel and its internals) and concrete within nuclear power plants can be disposed of as Class A waste, the least restrictive waste category specified in 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Wastes." In the one case where ^{137}Cs concentrations in the most contaminated concrete slightly exceeded the Class A waste limit, this concrete could still be disposed of as Class A waste by diluting it with less contaminated concrete. In PWR's, where the steam generators accounted for about 80 to 95% of the total residual radionuclides deposited outside of the pressure vessel, it was shown that entire steam generators could be disposed of as Class A waste with only one exception. In this case, the ^{63}Ni concentrations (μCi/gm) in the Rancho Seco steam generator were slightly above the Class A waste limit. A relatively simple decontamination operation would easily reduce the ^{63}Ni concentrations below Class A limits for these steam generators.

Radionuclide contamination of soils around the nuclear power plants was limited to small patches of very low concentrations of radionuclides. These areas were all within the inner security fences of the sites and were usually known locations where spills of radioactive material (mostly liquid samples) had occurred. Such areas included locations near condensate or borated water storage tanks, effluent sampling points, or equipment maintenance and/or cleaning areas. Generally, the radionuclide concentrations were below levels which would require remedial action. However, several small patches of soils and holding pond sediments contained radionuclide concentrations up to several tens to thousands of pCi/g of ^{60}Co plus ^{137}Cs. These soils would need to be disposed of as low level waste, but the volumes in most cases would be very small, e.g. tens of cubic meters.

The total residual radionuclide inventories (excluding the pressure vessel) at the seven nuclear plants examined in this study appear to be proportional to the product of the unit power level (megawattage) and the length of operations in years. Thus, extrapolations of the radionuclide inventories at other nuclear plants may perhaps be made. For the most part, the radionuclide compositions and inventories measured in this program were in reasonably good agreement with the limited data base used in the earlier conceptual assessment studies of the technology, safety, and costs of decommissioning a reference PWR (NUREG/CR-0130) and a reference BWR (NUREG/CR-0672). Thus, the conclusions reached in these conceptual studies from a radiological standpoint will essentially remain unchanged.

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1.0. INTRODUCTION

The U. S. Nuclear Regulatory Commission (NRC) has been charged with the responsibility of developing a general decommissioning policy for commercial nuclear facilities in the United States, including nuclear power plants. Since the nuclear industry has matured to the point where some of the early nuclear power plants have reached or will be nearing retirement status, it is imperative that the NRC develop detailed information to provide guidance for the decommissioning of these plants.

Several studies have conceptually assessed the technology, safety, and costs associated with various alternatives for decommissioning nuclear power plants.(1-5) The radiological assessments of the two reference reactors are contained in Appendix D-1. Actual decommissionings of several small reactors have also previously taken place. The radiological analyses available from these are reviewed and summarized in Appendix D-2. One of the key elements of such assessments is a characterization of the radionuclide inventory remaining within a nuclear power plant. This information is essential for: 1) understanding the radiological problems which will be encountered during decommissioning, 2) developing the most desirable decommissioning and waste handling methodologies for various decommissioning alternatives, and 3) gaining a better understanding of the long-term disposal problems potentially posed by the generated radioactive wastes. However, empirical data relating to the compositions, distributions, and quantities of residual radionuclides associated with piping, hardware, equipment, and concrete surfaces within nuclear power plants have been extremely meager, and theoretically calculated or estimated quantities have been used in most cases for previous decommissioning assessments.

To provide a more statistically valid data base for the residual radionuclide inventories in nuclear power stations, Pacific Northwest Laboratory was contracted to conduct an extensive sampling and measurements program at a number of nuclear power plants. This final report summarizes the results of the radionuclide characterization studies conducted at seven nuclear power stations from a decommissioning assessment viewpoint. Individual, detailed reports of the investigation at each station have previously been published.(6-12)

2.0 PROGRAM OBJECTIVES

This program was designed to provide NRC with a data base and decommissioning assessment of the residual radionuclide compositions, distributions, and quantities at commercial nuclear generating stations. Pacific Northwest Laboratory (PNL) has conducted a comprehensive sampling and analysis program at seven nuclear power stations. There have been four major objectives for the field sampling and analytical program.

The first major objective has been to provide an actual data base containing information regarding the range in compositions, quantities, and locations of radionuclide residues likely to be encountered in retired nuclear power stations. This has been accomplished using both older shutdown reactors closest to actual decommissioning, as well as newer generation, larger capacity reactor facilities. This research program has not dealt with the highly neutron activated components associated with the reactor pressure vessel and its internals, but rather has focused upon the residual radionuclides transported from the reactor vessel and deposited throughout the associated operating systems of the generating station. The program has also made measurements of residual radionuclides in contaminated concrete within the plants, and in soils and sediments in the immediate environs of the generating stations. A companion program conducted at PNL has assessed the problems posed to reactor decommissioning by the long-lived neutron activation products formed in reactor pressure vessel construction materials, especially the highly radioactive components inside the reactor pressure vessel.(13)

An important part of the first objective has been an evaluation of all significant radionuclides for decommissioning purposes. Emphasis has been placed not only on measuring the most abundant radionuclides shortly after shutdown (e.g., ^{51}Cr, ^{54}Mn, ^{55}Fe, ^{57}Co, ^{58}Co, ^{60}Co, ^{65}Zn, ^{90}Sr, ^{106}Ru, ^{110m}Ag, ^{134}Cs, ^{137}Cs, ^{141}Ce, ^{144}Ce) but also determining the abundances of radionuclides with extremely long half-lives, e.g., ^{59}Ni, ^{63}Ni, ^{94}Nb, ^{99}Tc, ^{129}I, ^{238}Pu, $^{239-240}$Pu, ^{241}Am, and ^{244}Cm, many of which are important for 10 CFR 61 waste classification purposes.(14) These radionuclides are of interest since they pose special environmental concerns from a long-term disposal standpoint. Several of these radionuclides have been noted from a theoretical basis to be potentially present in the neutron activated reactor materials, but their translocation from the reactor vessel and deposition throughout the other plant systems has not been examined previous to this study.

A second major objective has been to identify the origin of the residual radionuclide contamination and to correlate the observed radionuclide inventories with the reactor system construction materials, reactor operating history, and operational procedures. This is important since there has been an evolution in most areas of reactor construction and operating technology. For example, there have been very substantial changes in reactor construction materials from earlier generating stations to more recent ones. Thus, the residual radionuclide composition and inventory would be expected to reflect these differences.

The third goal of the research program has been to develop some degree of predictive capability so that generic assessments of residual radionuclide contamination in nuclear power stations can be performed in a more accurate manner than previous estimates. This has been accomplished through the field studies at seven different nuclear generating stations.

The final objective has been to provide an extensive data base which is applicable for use during formulation of policies and strategies for the decommissioning of retired nuclear power stations. At the present time several decommissioning alternatives are being considered. When the residual radionuclide inventory and compositions are better defined more informed decisions can be made concerning the optimal decommissioning alternatives and methodologies, as well as means for disposal of the generated wastes.

3.0 FIELD MEASUREMENTS PLAN

In order to achieve the above objectives, a field and analytical program was conducted at seven nuclear generating stations. The nuclear units sampled, the types of samples obtained, and the sampling procedures employed are briefly described in this section. The onsite sampling and measurements program, the laboratory preparation procedures, the radiochemical separation methods, and the radionuclide identification and quantification techniques are presented in detail in Appendix A.

3.1. CRITERIA FOR SELECTING NUCLEAR POWER STATIONS FOR EXAMINATION

Several factors influenced the selection of nuclear generating units for the field portion of this program. First, it was important that several nuclear reactors presently in shutdown or standby status be examined. These plants will probably be among the first to be decommissioned, so more extensive field sampling might be conducted which would be directly applicable to that process. Secondly, these older units have a longer operational history, thus allowing for increased production of long-lived radionuclides. In addition, the longer operational periods would allow greater opportunity for translocation and deposition of the radionuclides throughout the various plant systems. Substantial translocation might not be observed if sampling operations were conducted at the newer nuclear units. Four nuclear power stations were selected for examination which fell into the category of shutdown stations. These included the Pathfinder Generating Plant, the Humboldt Bay Nuclear Unit, Indian Point Station Unit One, and Dresden Nuclear Station Unit One.

Another factor influencing site selection was the desire to include nuclear power stations of differing types and designs, including both boiling water reactors (BWR's) and pressurized water reactors (PWR's). This factor was especially important for the operating units. Thus, the operating units chosen included the Monticello Nuclear Generating Plant (General Electric BWR), the Turkey Point Station Units 3 and 4 (Westinghouse PWR's), and the Rancho Seco Nuclear Generating Station (Babcock and Wilcox PWR). Each of the operating nuclear units was sampled during an extended outage. Sampling during these maintenance outages was advantageous since a wider variety of contaminated materials was available for sample acquisition purposes.

A final factor influencing our site selection was the cooperation and interest of the operating utility toward participation in the program. Participation by the operating utility entailed not only making available the nuclear facility for sampling but also a commitment of manpower before, during, and after our onsite activities. This involved pre-sampling discussions regarding the types and availability of samples for our program, assistance from operations and technical staff during sample procurement, escort and health physics coverage during our sampling activities, providing records and files regarding plant design and operational history, and review and comment on the topical report describing our activities at the site and the results of our radiological examination for the particular station.

In most instances after the goals and purposes of our research program were outlined and the potential benefits described, the utility personnel were receptive and every assistance was rendered during our onsite activities. Without the valuable assistance of the utility participants during our onsite sampling activities, and in reviewing our individual site reports, this program would not have been possible.

3.2. DESCRIPTION OF SELECTED NUCLEAR POWER PLANTS

Table 3.1 contains a listing of the nuclear units examined during this program, along with other pertinent information, including the operating utility, the unit generating capacity, reactor type and designer, startup and shutdown dates (as applicable), and the dates during which our onsite activities were conducted. A brief description of each nuclear power station is included below.

3.2.1. Pathfinder

The Pathfinder Generating Plant was a 66 MWe (203 MWt) boiling water type power station located 9 km northeast of Sioux Falls, South Dakota. Northern States Power Company is the owner and operator. Construction of the nuclear plant was completed in early 1964 and initial criticality was achieved on March 4, 1964. The plant was operated intermittently through a testing period of 42 months; and in September, 1967, was shut down due to failure of the steam separators within the reactor vessel. It was then decided to terminate the nuclear operations and convert the plant to a gas/oil fired unit. A partial decommissioning of the nuclear plant was conducted, including the removal of all fuel from the site and placing the nuclear plant into a mothballed state. The retrofitted fossil-fueled plant is still in use during periods of peak power demand.

3.2.2. Humboldt Bay

The Humboldt Bay Nuclear Unit is a General Electric designed, single cycle, internal natural circulation boiling water reactor rated at 50 MWe (165 MWt). The plant is near Eureka, California, and is operated by Pacific Gas and Electric Company. Construction was completed in late 1962, and initial criticality was achieved on February 16, 1963. The plant operated commercially from August 1, 1963 until July 2, 1976. It is currently in cold shutdown status by directive from NRC. The Humboldt Bay Nuclear Unit is currently being prepared for 30-year safe storage followed by dismantlement.

3.2.3. Indian Point Unit One

The Indian Point Station Unit One is located near Buchanan, New York, approximately 40 km north of New York City. Consolidated Edison Company of New York, Inc. operated Unit One, a pressurized water reactor rated at 285 MWe (615 MWt) from 1962 until 1974. The reactor was designed by Babcock and Wilcox and originally utilized enriched uranium and thorium oxide as fuel. After three years this core was replaced by a General Electric

TABLE 3.1. Nuclear Power Stations Examined

Station and Location	Utility	MWe	Reactor Type and Designer	Startup and Shutdown Dates		Date Sampled
Pathfinder, Sioux Falls, South Dakota	Northern States Power	58	BWR Allis Chalmers	1964	1967	May and June, 1980
Humboldt Bay, Eureka, California	Pacific Gas and Electric	63	BWR General Electric	1963	1976	April, 1981
Indian Point-1, Buchanan, New York	Consolidated Edison	265	PWR Babcock & Wilcox	1962	1974	May, 1982
Dresden-1, Morris, Illinois	Commonwealth Edison	210	BWR General Electric	1960	1978	August, 1982
Monticello, Monticello, Minnesota	Northern States Power	550	BWR General Electric	1972	Operating	April, May, Nov., 1981
Turkey Point, Homestead, Florida	Florida Power and Light	690	PWR Westinghouse	1972	Operating	October, 1981
Rancho Seco, Herald, California	Sacramento Public Utility District	930	PWR Babcock & Wilcox	1974	Operating	March, June, 1983

Company designed core which utilized slightly enriched uranium fuel. Another unique feature of the plant was the use of fossil-fired superheaters. The unit is currently idle in a controlled defueled state with its operating license revoked. Portions of its liquid rad-waste cleanup system are presently being shared with Unit 2.

3.2.4. Dresden Unit One

Dresden Nuclear Station Unit One is a dual cycle boiling water reactor located near Morris, Illinois, approximately 100 km southwest of Chicago. The nuclear unit was designed by General Electric and is rated at 210 MWe (700 MWt). The unit operated commercially from July, 1959, through October, 1978, at which time it was placed into cold shutdown. Commonwealth Edison, the owner and operator, is currently proceeding with 30-year safe storage.

3.2.5. Monticello

The Monticello Nuclear Generating Plant is located at Monticello, Minnesota, some 60 km northwest of Minneapolis. The station contains a single unit 545 MWe (1670 MWt), boiling water reactor designed by General Electric. The plant has been in commercial operation since early 1971, and is owned and operated by Northern States Power.

3.2.6. Turkey Point Units 3 and 4

The Turkey Point Station Units 3 and 4 nuclear plants are located adjacent to two oil and gas-fired units on the shore of Biscayne Bay approximately 40 km south of Miami, Florida. The station is owned and operated by Florida Power and Light. The nuclear units are Westinghouse designed pressurized water reactors rated at 760 MWe (2200 MWt), which reached operational status within one year of each other. Initial criticality was achieved at Unit 3 on October 20, 1972, and at Unit 4 on June 11, 1973. Both units reached rated power in March, 1974, and have operated commercially since 1974.

3.2.7. Rancho Seco

The Rancho Seco Nuclear Generating Station contains a single pressurized water reactor of Babcock and Wilcox design. The generating station is located approximately 40 km southwest of Sacramento, California, and is operated by the Sacramento Municipal Utility District (SMUD). The unit is rated at 935 MWe (2770 MWt) and began commercial power operations in late 1974.

3.3. SAMPLING AND ANALYSIS OF CONTAMINATED MATERIALS

The details of the sampling and analysis procedures, employed both at the nuclear power plants and at our laboratory, are provided in Appendix A. Appendix B contains an inventory of the wide variety of samples collected at the plants during this study. A summary of the sample types collected for radiochemical analyses during this program is given in Table 3.2.

TABLE 3.2. Sample Inventory from the Nuclear Power Stations Examined

	Number of Samples				
Station	Piping & Hardware	Corrosion Film Scrapings	Concrete Cores	Misc.	Soil
Pathfinder	52	0	22	6	10
Humboldt Bay	5	12	27	10	16
Indian Point-1	8	10	15	3	0
Dresden-1	2	12	8	2	8
Monticello	9	2	10	1	4
Turkey Point	4	6	16	2	11
Rancho Seco	4	5	2	1	8

Briefly, the objective of the sampling program was to obtain as many samples of opportunity as possible of contaminated piping, hardware, equipment, concrete, and soils which could be analyzed to provide accurate measurements of the residual radionuclide concentrations associated with these materials. Components from both the primary and secondary coolant systems were procured when possible, along with materials associated with the rad-waste systems and other contaminated portions of the plants. Concrete cores were taken from areas of the plants having some of the highest contamination levels on the floors. Cores were also collected from other potentially contaminated areas in the reactor buildings, turbine buildings, auxilliary buildings, and any other structures reported by the utilities to be contaminated. Soil samples were normally collected in the four compass directions adjacent to the plants, and from known locations of contamination, such as liquid spill areas in tank yards.

Samples were initially analyzed by gamma-ray spectrometry at the reactor sites, and then subjected to comprehensive quantitative radiochemical analyses at our laboratory in order to measure all important radionuclides from a decommissioning standpoint. Emphasis was placed on measuring those long-lived radionuclides which would be associated with the residual contamination for tens to thousands of years, e.g., ^{55}Fe, ^{60}Co, ^{59,63}Ni, ^{90}Sr, ^{94}Nb, ^{99}Tc, ^{129}I, ^{134,137}Cs, ^{152,154}Eu, and isotopes of Pu, Am, and Cm. The concentrations of these, and all other shorter-lived radionuclides detected by gamma-ray spectrometry, have been reported as microcuries/cm^2 or microcuries/gm of contaminated material, so that estimates of the curie contents of the various contaminated systems could be made.

4.0. RESULTS AND DISCUSSION

The results of the sampling and analytical program for the seven reactor sites examined in this program indicate a wide range exists in: 1) the relative radionuclide composition of contaminated materials at the reactor sites, 2) the distribution of the residual radionuclide inventory throughout the generating stations, and 3) the total inventory present. The observations resulting from the field and analytical program are discussed below; are compared to previous assessments; and are related, where possible, to site operating parameters and history. A more complete discussion of the inventory construction process is presented in Section 4.2. The inventories estimated for the seven plants are based upon a limited number of samples and data points. The greatest uncertainty in these inventory estimates is associated with the sampling process; if large variations in radioactive corrosion product deposition existed throughout plant systems, extrapolations from our limited number of samples would be subject to corresponding uncertainties. Thus, the inventory estimates presented herein should be used with some degree of caution. We feel that our inventory estimates are accurate to within ±50%, based upon comparisons with other inventory studies.(6,10)

4.1. RESIDUAL RADIONUCLIDE COMPOSITIONS

The relative radionuclide composition of contaminated materials observed at the seven reactor sites ranged considerably. The range in compositions was influenced by numerous factors including: 1) the elapsed time from the last reactor operations; 2) rated generating capacity; 3) materials of construction for the operating systems; 4) reactor type e.g. PWR, BWR, dual cycle; 5) coolant chemistry and corrosion control; 6) fuel integrity during operations; 7) episodic equipment failure and leakage of contaminated liquids; and 8) the specific source for the samples, e.g. primary system, secondary or steam system, rad waste system, or the spent fuel storage facility.

4.1.1 Piping and Hardware - Generic Observations

The relative radionuclide compositions for the total plant inventories of contaminated piping, equipment, and hardware estimated for the seven sites sampled are shown in Table 4.1. These inventories include only the radioactive contamination deposited on corrosion film and crud surfaces of the various plant systems, and do not include the highly activated components of the pressure vessel.

The most abundant radionuclides in samples two to three months old or older generally included ^{54}Mn, ^{55}Fe, ^{58}Co, ^{60}Co, and ^{63}Ni. Zinc-65 was present in relatively high concentrations in BWR corrosion film samples. Traces of transuranic radionuclides, including ^{238}Pu, $^{239-240}$Pu, ^{241}Am, ^{242}Cm, and ^{244}Cm, were also observed in the residues. Other long-lived radionuclides of interest from a waste management standpoint, such as ^{94}Nb, ^{99}Tc, and ^{129}I, were frequently below their limits of detection; and ^{90}Sr was always present in extremely low concentrations. The ^{90}Sr,

TABLE 4.1. Long-lived Residual Radionuclide Compositions in Total Plant Inventories* at Seven Nuclear Generating Stations

	Composition in Percent of Total Activity Decay Corrected to Shutdown Date or Sampling Date						
Radionuclide	Pathfinder	Humboldt Bay	Dresden-1	Monticello	Indian Point-1	Turkey Point-3	Rancho Seco
Mn-54	--	3	0.9	1	4	0.4	4
Fe-55	0.36	90	28	1	67	31	28
Co-57	--	--	--	--	--	43	24
Co-60	1.0	6	46	11	15	24	18
Ni-59	2×10^{-4}	--	0.09	--	0.02	0.004	0.1
Ni-63	0.03	0.2	5	0.04	2	0.1	19
Zn-65	98.6**	--	19	84	11	1	0.09
Sr-90	$<1 \times 10^{-4}$	0.004	0.007	0.002	0.0007	0.0008	<0.01
Nb-94	$<3 \times 10^{-6}$	<0.004	<0.003	<0.1	0.0008	<0.004	<0.004
Tc-99	$<1 \times 10^{-5}$	3×10^{-4}	4×10^{-5}	8×10^{-5}	8×10^{-5}	0.008	<0.005
Ag-110m	--	--	--	--	--	--	4
I-129	--	$<3 \times 10^{-6}$	$<1 \times 10^{-5}$	$<1 \times 10^{-6}$	2×10^{-5}	<0.003	$<1 \times 10^{-5}$
Cs-137	3×10^{-4}	0.5	0.04	2	0.5	--	0.4
Ce-144	--	--	1	--	--	0.2	<0.04
TRU***	3×10^{-6}	0.005	0.1	0.008	0.002	0.006	0.001
Total Plant Inventory (Curies)	57(a)	596(b)	2350(c)	448(d)	1070(e)	2580(f)	4460(g)

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*Excludes highly activated metal components of the reactor pressure vessel and internals, contaminated concrete, and soils.

**At time of station shutdown, ^{65}Zn was the overwhelming contributor to the total residual radionuclide inventory at Pathfinder, but had all decayed during the 13 years between shutdown and our sampling and analysis program. The relative radionuclide composition and total inventory in this table was reconstructed from our data and a measured ^{65}Zn/^{60}Co ratio made by NSP shortly after shutdown.

***Transuranic alpha-emitting radionuclides with half-lives greater than 5 years, including ^{238,239,240}Pu, ^{241}Am, ^{243}Am, and ^{244}Cm.

(a) on September, 1967
(b) on July, 1976
(c) on October, 1978
(d) on November, 1981; includes 1% ^{103}Ru
(e) on October, 1974
(f) on October, 1981
(g) on June, 1983; also contained 1.8% ^{125}Sb

^{99}Tc, and ^{129}I were generally very soluble and did not accumulate in any significant degree in corrosion product deposits. Although ^{94}Nb may be produced in relatively large quantities in the reactor vessel by activation of trace niobium impurities in stainless steel pressure vessel components(1,2,13), its concentrations in corrosion product deposits were extremely low. This may be the result of the extreme insolubility of niobium. Occasionally, radionuclides such as ^{103,106}Ru, $^{141-144}$Ce, and ^{110m}Ag were detectable in the corrosion films.

As shown in the table, the relative radionuclide composition was quite variable. However, ^{55}Fe and ^{60}Co were the two most abundant radionuclides in all cases except Monticello and Pathfinder. These two radionuclides constituted over 95% of the estimated inventories at Humboldt Bay and Turkey Point. At Indian Point Unit One, Dresden Unit One, and Rancho Seco they accounted for 86, 84, and 60%, respectively, of the total estimated inventory. Although ^{55}Fe and ^{60}Co accounted for the majority of the inventory (greater than 60% at five of the seven stations), the relationship between the two radionuclides was quite variable. The ratio of ^{55}Fe to ^{60}Co at the six generating stations where ^{55}Fe and ^{60}Co constituted the majority of the inventory ranged from 15 to 1 at Humboldt Bay to 0.36 to 1 at Pathfinder. At Monticello, where ^{65}Zn constituted 90% of the total inventory, the ^{55}Fe to ^{60}Co ratio was even lower, 0.09 to 1. This large variability was presumably due to differences in: 1) the parent element composition of construction materials used in the pressure vessel and primary coolant loop, 2) differences in the water chemistry which controlled the corrosion and deposition of these radionuclides, and 3) differences in operating history which affect the production ratios since the radionuclides have an approximate factor of two difference in half-life.

The transuranic radionuclides (^{238}Pu, $^{239-240}$Pu, ^{241}Am, ^{242}Cm, and ^{244}Cm) constituted percentages of the total inventory ranging from 3 X 10^{-6}% at Pathfinder to 0.1% at Dresden Unit One. The value observed at Pathfinder was somewhat anomalous due to its short operational period prior to shutdown, along with the fact that there were no detected fuel failures during this brief period. Considering the other six stations, the total transuranic component of these inventories ranged a factor of fifty, from 0.002% at Indian Point Unit One to 0.1% at Dresden Unit One.

The largest ranges, as a percentage of the total inventory, were noted for ^{65}Zn and ^{63}Ni, which showed ranges of 1090 and 630, respectively. This wide range was related to the composition of the materials of construction used in the primary systems of the reactors. The large amounts of ^{65}Zn observed at Monticello, Dresden Unit One, and Indian Point Unit One were the result of the use of admiralty brass heat exchangers (29% zinc). These were replaced with stainless steel at Dresden Unit One, but a large residual corrosion product inventory was still present in the operating systems at the time of our sampling. Pathfinder also utilized admiralty brass heat exchangers, and at the time of shutdown ^{65}Zn accounted for over 98% of the total radioactivity. Humboldt Bay power plant also initially utilized admiralty brass heat exchangers which resulted in generating relatively large amounts of ^{65}Zn during early reactor operations.

The wide range in ^{63}Ni was due to a relatively high percentage abundance observed at Rancho Seco (19%) and an unusually low abundance at Monticello (0.04%). The large component of ^{63}Ni in the Rancho Seco inventory was due to the more extensive use of inconel (80% nickel) in the primary system of this reactor. The low percentage of ^{63}Ni at Monticello was typical of newer BWRs which make minimal use of nickel alloys since they are subject to higher rates of corrosion in the relatively more oxidizing environment of the BWR primary coolant loop. Excluding Rancho Seco and Monticello, the relative abundance of ^{63}Ni ranged a factor of 50, from 0.1% of the total inventory at Turkey Point Unit-3 to 5% at Dresden Unit One.

The highest concentrations of radionuclides in corrosion films were observed in the systems exposed to the primary coolant. Table 4.2 lists the concentration ranges and average concentrations for a number of the radionuclides observed in corrosion films attached to piping from primary coolant systems. These data were synthesized from individual measurements on appropriate samples given in Appendix C. Large variations in residual radionuclide concentrations on the primary coolant piping existed between the various stations, so the average value presented in the table should be considered with appropriate reservations.

The residual concentrations of fission products, such as ^{90}Sr and ^{137}Cs and the transuranic radionuclides (Pu, Am, Cm), on piping are directly related to the fuel integrity during the operating history of the plant. At power stations where fuel element failures were relatively frequent or severe (e.g. Humboldt Bay, Dresden Unit One), the concentrations of fission products and transuranic radionuclides in the corrosion films were higher than observed at stations where the fuel remained relatively free from failures.

Secondary coolant loops in PWRs and condensate systems in BWRs contained much lower radionuclide concentrations than observed in primary loop or feedwater samples. Typically, radionuclide concentrations were approximately two orders of magnitude lower in the secondary system samples. Table 4.3 contains concentration ranges and average concentrations observed in the corrosion films from nonprimary loop samples. The actual measurements from which this table was constructed are given in Appendix C. The average concentration values presented in the table should be used with reservations since the ranges shown in the table typically cover several orders of magnitude.

The piping and hardware data from the seven sites, based upon reactor type, are discussed in detail below.

4.1.2. PWR Piping and Hardware

Contaminated piping and hardware samples were obtained from three PWRs-Indian Point Station Unit One, Rancho Seco Nuclear Generating Station, and Turkey Point Station Units 3 and 4. Indian Point Station Unit One is an early design Babcock & Wilcox reactor, Rancho Seco is a more recent

TABLE 4.2. Concentration Ranges of Radionuclides in Corrosion Films on Piping Exposed to Primary Reactor Coolant[(a)]

Radionuclide	Halflife (yr)	Concentration Range (μCi/cm^2)	Average Concentration (μCi/cm^2)
Mn-54	0.854	0.028 - 4.4	1.2
Fe-55	2.7	0.039 - 149	33
Co-60	5.27	0.16 - 23	6.4
Ni-59	75,000	$<5 \times 10^{-5}$ - 2.6×10^{-2}	6.6×10^{-3}
Ni-63	100	0.003 - 1.3	0.80
Zn-65	0.67	0.0005 - 5.8	5.6(b)
Sr-90	28.5	$<3 \times 10^{-5}$ - 8.4×10^{-3}	3.2×10^{-3}
Nb-94	20,000	$<1 \times 10^{-5}$ - 5.0×10^{-4}	2.2×10^{-4}
Tc-99	2.13×10^{5}	4.5×10^{-6} - 5.6×10^{-4}	1.5×10^{-4}
I-129	1.57×10^{7}	$<1 \times 10^{-6}$ - 4.3×10^{-6}	1.4×10^{-6}
Cs-134	2.06	0.019 - 0.046	0.030
Cs-137	30.2	0.003 - 0.17	0.056
Pu-238	87.8	2.4×10^{-6} - 4.5×10^{-3}	1.1×10^{-3}
Pu-239,240	24,400	1.5×10^{-6} - 4.4×10^{-3}	9.8×10^{-4}
Am-241	433	1.8×10^{-6} - 8.3×10^{-3}	1.8×10^{-3}
Cm-242	0.447	7.2×10^{-6} - 2.4×10^{-1}	7.2×10^{-2}
Cm-244	18.1	2.2×10^{-6} - 2.5×10^{-3}	6.8×10^{-4}

(a) Decay corrected to date of shutdown for terminated reactors or sampling date for operating units. Excludes data from Pathfinder Generating Plant, which was atypically low.

(b) Average value for two BWR units.

TABLE 4.3. Concentration Ranges of Radionuclides in Corrosion Films Internally Deposited in Piping and Hardware Exposed to Liquid Radwastes and Secondary Coolant

Radionuclide	Halflife (yr)	Concentration Range (pCi/cm^2)	Average Concentration (pCi/cm^2)	
Mn-54	0.854	2 - 4.70×10^5	160,000	(3)[a]
Fe-55	2.7	710 - 7.1×10^6	1.0×10^6	(7)
Co-60	5.27	64 - 2.74×10^5	42,000	(7)
Ni-59	75,000	0.63 - 15	6.2	(5)
Ni-63	100	3 - 10,000	1,900	(6)
Zn-65	0.67	<0.4 - 27,400	14,600	(2)[b]
Sr-90	28.5	<0.009 - 260	88	(3)
Nb-94	20,000	<0.1 - 100	--	
Tc-99	2.13×10^5	<0.05 - 0.2	--	
Sb-125	2.77	9.3 - 1,200	420	(3)
I-129	1.57×10^7	<0.0006 - 0.9	--	
Cs-134	2.06	<0.3 - 1,900	--	
Cs-137	30.2	<0.6 - 4,000	860	(5)
Ce-144	0.78	<0.6 - 2,000	--	
Pu-238	87.8	0.0014 - 51	7.4	(7)
Pu-239,240	24,400	0.0012 - 24	3.6	(7)
Am-241	433	0.0009 - 41	7.3	(6)
Cm-242	0.447	0.0013 - 3,600	820	(5)
Cm-244	18.1	0.0015 - 58	12	(5)

a) Numbers in parentheses indicate the number of reactor sites used for construction of the average value presented.
b) Average value of two BWR units.

Babcock & Wilcox unit, and Turkey Point Units 3 and 4 are Westinghouse design reactors.

4.1.2.1. Indian Point Unit One. Twenty-four hardware and scraping samples were obtained from Indian Point Unit One. The detailed radioanalytical measurements of these samples are given in Appendix C. The highest radionuclide concentrations were found associated with the primary coolant systems, followed by the liquid rad-waste system, seal water and primary makeup water system, fuel storage basin, main steam line, and the condenser and condensate system. The most radioactive sample was a scraping from the primary side of one of the nuclear boilers (boiler No. 14). The most abundant radionuclides at the time of sampling, in order, were ^{55}Fe, ^{60}Co, ^{63}Ni, ^{137}Cs, and ^{59}Ni. Although ^{65}Zn and ^{54}Mn were minor constituents of the residual radioactivity at the time of our on-site sampling, at the time of shutdown they were the third and fourth most abundant radionuclides with half-lives greater than 245 days. This sample provided the best estimate of the residual radionuclide concentrations in the corrosion film surfaces of the systems exposed to the primary coolant.

For the seventeen scraping samples obtained at Indian Point, the ^{134}Cs/^{137}Cs ratio (corrected for decay to the shutdown date) ranged from 0.31 to 2.5 and averaged 0.73 ± 0.31. The considerable variation indicates that the radiocesium contamination had occurred at different time periods for the various samples.

Selective deposition of the various radionuclides occurred during transport and cycling throughout the different plant systems. For example, the ^{137}Cs/^{60}Co ratio was quite variable in the suite of samples from Indian Point Unit One, ranging from 0.024 to 20. These ratios appeared to cluster into three groups. The first group was typified by ratios much greater than 1.0 (5 to 16). These samples included seal water lines and other systems in contact with relatively uncontaminated waters. The second cluster of ratios was centered about 1.0, and included most general area scrapings, the fuel pool area, and a structural steel sample from below the reactor vessel. The final set of ratios was much less than 1, and was typified by the nuclear boiler sample which had a ratio of 0.02. These observations indicate that ^{60}Co deposition is more rapid compared to ^{137}Cs, and the greater the transport or recycling of the contaminated liquids, the higher the ^{137}Cs/^{60}Co will be in the corrosion films.

The ^{238}Pu/^{60}Co and $^{239-240}$Pu/^{60}Co ratios for the Indian Point Unit One hardware samples were much more consistent, except for one sample. The ratios averaged (± one sigma) $3.2 \pm 2.3 \times 10^{-5}$ for ^{238}Pu and $5.0 \pm 5.0 \times 10^{-5}$ for $^{239-240}$Pu. The ^{238}Pu/^{137}Cs and $^{239-240}$Pu/^{137}Cs ratios were not nearly as consistent as the Pu/^{60}Co ratios. For the six hardware samples analyzed radiochemically, the ratios ranged over five orders of magnitude, from 2.3×10^{-8} to 2×10^{-3}. Thus, it appears the transuranium radionuclides were deposited into the corrosion films in a manner somewhat similar to ^{60}Co, since the ratio to ^{60}Co was relatively consistent, while the cesium was deposited in an irregular fashion relative to ^{60}Co or plutonium.

The secondary loop at Indian Point Unit One contained relatively high levels of contamination compared to Turkey Point and Rancho Seco. Considerable leakage of the primary coolant into the secondary loop had occurred during the operating history at Indian Point Unit One, resulting in this contamination.

4.1.2.2. Turkey Point Units 3 and 4. At Turkey Point, nine corrosion film and hardware samples were obtained. Detailed radiochemical analyses are given in Appendix C. Three of the samples were related to primary coolant systems. These included an alumina grit sample from the steam generator primary side decontamination process, and two scrapings from the steam generator bowls after decontamination. The $^{137}Cs/^{60}Co$ ratios were extremely variable, ranging from 0.0077 to 3.8. Because of the very low radiocesium content in the corrosion films, $^{134}Cs/^{137}Cs$ ratios could not be calculated. The $^{238}Pu/^{60}Co$ ratios in five of the six samples analyzed radiochemically averaged $3.3 \pm 1.4 \times 10^{-5}$. If the sixth sample, with a somewhat higher ratio is included, the ratio was $3.8 \pm 2.2 \times 10^{-5}$. As was the case at Indian Point Unit One, the $Pu/^{60}Co$ ratio was much more consistent than $^{137}Cs/^{60}Co$. The residual radionuclide concentrations associated with the corrosion films on secondary loop surfaces were extremely low, indicating very little leakage from the primary loop.

4.1.2.3 Rancho Seco Nuclear Generating Station. Ten scraping and hardware samples were obtained at Rancho Seco. Detailed radiochemical analyses are given in Appendix C. Four were associated with the primary system, and the other six were from secondary components or scrapings of boron residues from leakages of slightly contaminated liquids. The $^{137}Cs/^{60}Co$ ratios were quite variable for the seven samples in which both ^{60}Co and ^{137}Cs were detectable. The ratios ranged from 0.006 to 10. The high ratios were both associated with boric acid residues associated with primary coolant leakage (0.63 and 10), while all other primary and secondary corrosion films contained ratios much less than 1.0. The $^{134}Cs/^{137}Cs$ ratio, where both radionuclides were detectable, were considerably greater than at Indian Point One. The two samples of primary system corrosion film had ratios of 0.21 and 0.22, while the four secondary samples ranged from 0.48 to 1.2, and averaged 0.68 ± 0.35. The $^{238}Pu/^{60}Co$ ratio for the five samples analyzed radiochemically was $3.7 \pm 4.2 \times 10^{-5}$; again, much more consistent than $^{137}Cs/^{60}Co$ ratios. The secondary loop contamination, as also observed at Turkey Point, was very minimal.

4.1.2.4. Summary of PWR Piping and Corrosion Films. The radionuclide contents of the corrosion films from the three PWRs sampled were quite variable in composition. Ratios of $^{134}Cs/^{137}Cs$ were variable from reactor to reactor, but more consistent within a reactor site. Ratios of $^{137}Cs/^{60}Co$ were extremely variable from system to system within a reactor site. The $^{238}Pu/^{60}Co$ ratios were quite consistent both within components of a specific reactor site and also from site to site. Based upon sixteen samples of corrosion films from both primary and secondary systems, the average ratio for $^{238}Pu/^{60}Co$ was $3.6 \pm 2.8 \times 10^{-5}$. This rather consistent ratio would indicate that the plutonium is incorporated or substituted into the primary loop corrosion films in a manner similar to that for ^{60}Co.

4.1.3. BWR Piping and Hardware

Four BWR's were sampled, including Humboldt Bay, Dresden Station Unit One, Pathfinder, and Monticello. Pathfinder has already been partially decommissioned and converted to a fossil fueled unit. Both Dresden Unit One and the Humboldt Bay nuclear unit are older BWR's which are currently shut down and may be decommissioned in the near future. Monticello is a newer, larger BWR and is more typical of present generation units.

4.1.3.1. Humboldt Bay. Detailed radiochemical analyses of the Humboldt Bay samples are given in Appendix C. Humboldt Bay piping and hardware samples contained higher ^{55}Fe concentrations, relative to other radionuclides, than observed at other sites. The ^{134}Cs/^{137}Cs ratios observed in Humboldt Bay components ranged from 0.08 to 2.4. The ^{137}Cs/^{60}Co ratios were also extremely variable, ranging from 0.0001 to 110. The highest ratios were associated with the fuel pool area samples and the offgas system samples. This was expected since the radiocesium might well be enhanced in the fuel pool areas due to leakage from failed fuel elements stored in the pool, and in the offgas system due to leakage of the gaseous precursor of ^{137}Cs, ^{137}Xe. The components associated with feedwater or steam systems had lower ^{137}Cs/^{60}Co ratios, typically approximately 0.002, and ranging from 0.0013 to 0.010. The ^{238}Pu/^{60}Co ratios at Humboldt Bay ranged from 0.00011 to 0.0091, and averaged $3.3 \pm 2.9 \times 10^{-4}$ for the thirteen samples analyzed radiochemically. This ratio is approximately a factor of ten higher than observed for the three PWR's, and is due to the fact that extensive fuel element failures occurred in the early operations of the Humboldt Bay reactor.

4.1.3.2. Dresden Unit One. As was the case at Humboldt Bay, Dresden One had been in shutdown status prior to our sampling (3.8 yrs). Residual radionuclide levels in contaminated piping and hardware at Dresden Unit One were dominated by ^{60}Co, followed by ^{55}Fe, ^{65}Zn and ^{63}Ni. Long-lived fission product concentrations were lower at Dresden compared to Humboldt Bay. At Dresden One the ^{134}Cs/^{137}Cs ratios were more consistent, ranging from 0.25 to 0.46 and averaging 0.38 ± 0.08. The ^{137}Cs/^{60}Co ratios at Dresden One ranged over three orders of magnitude. High ratios (0.1 to 0.5) were associated with the fuel pool system and the fuel canal, as observed at other plants. Lower ratios, 0.0005 to 0.005, were associated with the steam and turbine systems. The ^{238}Pu/^{60}Co ratios observed at Dresden One, for the six samples analyzed radiochemically, averaged $9.1 \pm 5.8 \times 10^{-5}$. This average ratio was a factor of 3 lower than observed at Humboldt Bay and within approximately a factor of 2 of the average ratio for the three PWR's sampled.

4.1.3.3. Monticello. The Monticello station is a relatively new BWR, having operated approximately 10 years at the date of our sampling. The residual radionuclide levels in contaminated piping and hardware were dominated by ^{65}Zn, ^{60}Co, and ^{137}Cs. Relatively higher concentrations of fission products were observed in the corrosion deposits compared to Humboldt Bay and Dresden. The piping and hardware corrosion films at Monticello contained $^{134}Cs/^{137}Cs$ ratios which ranged from 0.01 to 0.16 and averaged 0.08 ± 0.05 for the seven samples in which both cesium radionuclides were detected. The $^{137}Cs/^{60}Co$ ratios ranged from 125 in the offgas system to 0.012 in the condensate demineralizer filter. The $^{238}Pu/^{60}Co$ ratios for the seven samples in which ^{238}Pu was present in detectable concentrations averaged $7.7 \pm 11.8 \times 10^{-5}$. This ratio is very similar to that observed at Dresden and within approximately a factor of 2 of that for the three PWR's.

4.1.3.4. Pathfinder. The residual radionuclide composition of contaminated corrosion films at Pathfinder were uniquely different due to its shorter operating history and the fact that no fuel element failures occurred during its operation. The spectrum of radionuclides was therefore dominated by activation products, and extremely low concentrations of fission products and transuranic radionuclides were present. At shutdown, the most abundant radionuclide, by far, was ^{65}Zn, which originated from neutron activation of zinc corrosion products leached into the primary coolant from the admiralty heat exchangers in the condenser. At the time of our sampling (13 years after shutdown), the ^{65}Zn had decayed away and the residual radionuclide composition was primarily due to ^{60}Co, with tenfold lower concentrations of ^{63}Ni and ^{55}Fe.

4.1.3.5. Summary of BWR Piping and Hardware. The ratios of $^{134}Cs/^{137}Cs$ were variable at the three BWR's sampled, both from system to system within a site and between sites. Some of the variation was due to uncertainties associated with the date of deposition of the radionuclides, since two of the reactors had been shut down for several years prior to our sampling. The $^{137}Cs/^{60}Co$ ratios were also variable from system to system at a given site. Higher ratios were associated with the offgas systems and the fuel storage pool areas, where one might expect radiocesium concentrations to be elevated compared to ^{60}Co. Lower ratios were observed in the steam and turbine systems. The $^{238}Pu/^{60}Co$ ratios at the three BWR's were higher than observed at the PWR sites, 1.3×10^{-4} for 25 BWR samples versus 3.6×10^{-5} for the PWR samples.

4.1.4 Contamination Residues in Concrete

Radionuclide contamination of concrete in nuclear power plants was of two types: 1) surficial contamination resulting both from spills of radioactive materials and/or deposition of radioactive aerosols and 2) contamination produced in-situ by neutron activation of concrete near the pressure vessel. Concrete areas subjected to neutron exposure within the plants were limited primarily to the bioshield and the sump area directly beneath the reactor vessel. Surficial contamination, although patchy in nature, was more widespread throughout the plants but generally limited to areas of spills or to high radiation areas where maintenance work had been conducted. Table 4.4 contains the observed ranges of radionuclide concentrations and an average concentration associated with concrete surfaces for a number of long-lived radionuclides. The concrete cores selected for inclusion in this table were the most highly contaminated from each site. Thus, these data represented the worst cases en-

TABLE 4.4. Concentration Ranges of Radionuclides Associated with Concrete from Highly Contaminated Areas Within Selected Nuclear Generating Stations

Radionuclide	Halflife (yr)	Concentration Range (pCi/cm^2)	Average Concentration (pCi/cm^2)	
Mn-54	0.854	35 - 21,000	6,200	(5)[a]
Fe-55	2.7	2,200 - 830,000	200,000	(5)
Co-60	5.27	590 - 460,000	110,000	(5)
Ni-59	75,000	30 - 2,400	860	(3)
Ni-63	100	3,100 - 6,400	4,800	(2)
Sr-90	28.5	1.6 - 480	170	(4)
Nb-94	20,000	<3 - 50	--	
Tc-99	2.13×10^5	0.27 - 2.4	1.6	(3)
Ru-106	1.0	<30 - 190	--	
Ag-110m	0.686	59 - 3,600	1,800	(2)
Cs-134	2.06	70 - 1.7×10^6	310,000	(6)
Cs-137	30.2	550 - 2.0×10^6	370,000	(6)
Ce-144	0.78	26 - 3.1×10^6	620,000	(5)
Eu-152	12.4	9 - 3,100	1,000	(3)
Eu-154	8.5	90 - 1,500	680	(3)
Eu-155	4.96	10 - 500	260	(2)
Pu-238	87.8	0.025 - 48	14	(4)
Pu-239,240	24,400	0.089 - 21	7.7	(4)
Am-241	433	0.10 - 30	8.7	(4)
Cm-242	0.447	0.06 - 1,800	880	(3)
Cm-244	18.1	0.05 - 52	13	(4)
Np-237	2.14×10^6	0.013 - 0.026	0.016	(3)

a) Number of reactor units included to calculate the average value.

countered in the concrete sampling program. These locations were typically from the reactor sump area or a high radiation area associated with some component of the primary coolant system or rad-waste system. As with the previous two tables showing ranges and average values, the average concentrations presented here for concrete contamination should be used with appropriate reservations since the ranges encountered were so large. Detailed measurements of the radionuclide concentrations in all of the concrete core samples from the seven nuclear power stations examined are given in Appendix C.

Where the surface of the concrete had been kept well coated with paint or epoxy, the contamination resided mainly on the coating surface, and the majority of the radionuclides could be removed by stripping the coating material, as illustrated in Table 4.5. As shown in the table, 46 to 99.8% of the ^{60}Co could be completely removed from the concrete surfaces of these cores by stripping the epoxy or gray paint coatings. The clear silicate sealer was very difficult to remove and retained most of the ^{60}Co contamination on the concrete. Where surface scratches or cracks in the concrete have developed, the radionuclides penetrate to deeper levels and could not be completely removed by stripping the partially painted surfaces. Even then, the decrease in radionuclide concentrations with depth in the concrete is very pronounced. As shown in Figure 4.1, the decrease in ^{137}Cs concentration with depth for three concrete cores for the sump floor of the reactor building at Indian Point Unit One amounted to approximately three orders of magnitude between the surface and a depth of 1 cm. The increase in concentration of ^{137}Cs at a depth of approximately 4 cm is associated with an interface between two layers of concrete poured at different times. The top layer easily separated from the rest of the core during the coring process. Apparently, a small amount of contaminated liquid migrated vertically to this layer and then spread out horizontally at the interface. This subsurface maximum could not be the result of thermalization of neutrons escaping the pressure vessel and fissioning the trace uranium impurities in the concrete. Neutron thermalization and uranium activation should occur over a broader region and extend to a greater depth than observed in this core.

Cesium-137 and ^{134}Cs are preferentially sorbed onto bare concrete relative to other radionuclides since cesium has the ability to undergo ion exchange onto minerals associated with the concrete matrix. The preferential adsorption of radiocesium was only noted for uncoated concrete or surfaces which had lost integrity in the paint coating. Operating nuclear power plants would be wise to insure that paint coatings on concrete surfaces exposed to potential radionuclide contamination be carefully maintained to minimize subsurface contamination of the concrete.

The concrete cores collected directly under the pressure vessel showed the effects of both surface contamination and subsurface neutron activation of stable elements present in the concrete. Figure 4.2 shows the depth distribution of ^{60}Co in cores collected under the pressure vessels at the Pathfinder, Indian Point Unit One, and Turkey Point Unit 3 reactors. The initial rapid decrease in ^{60}Co concentration with depth

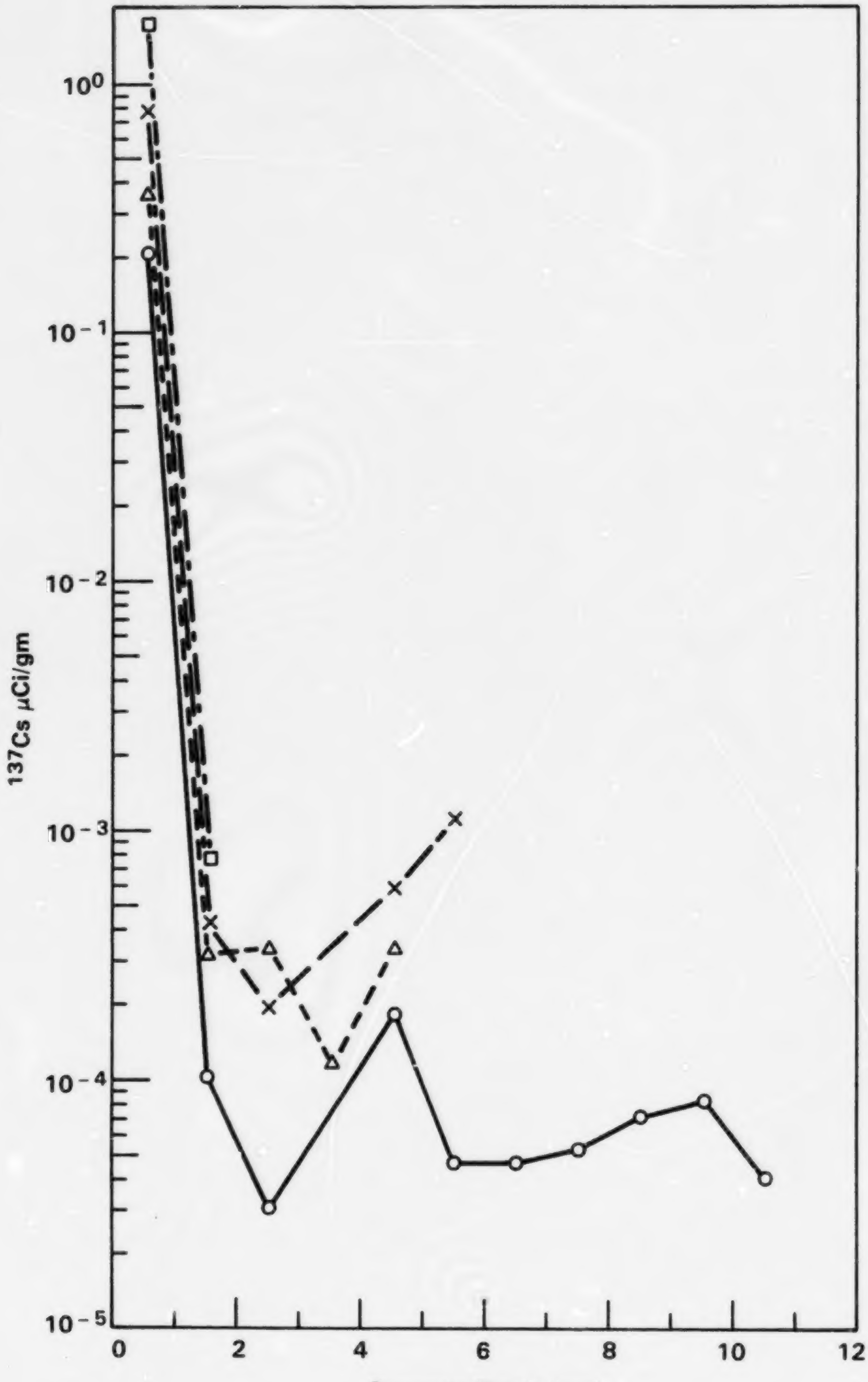

FIGURE 4.1. Cesium-137 Depth Distribution in Concrete Cores from the -5' Level of the Reactor Building at Indian Point-1, May 1982

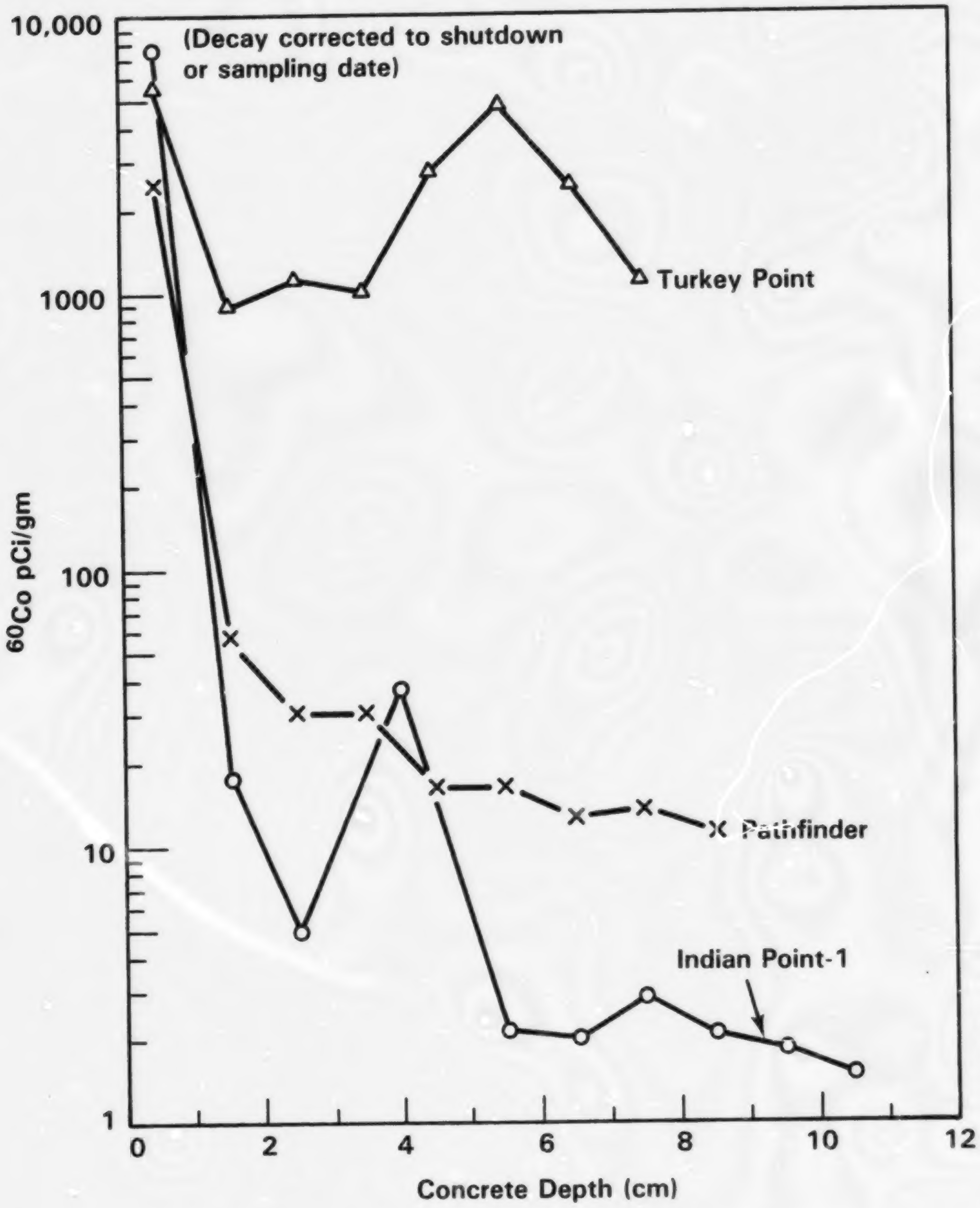

FIGURE 4.2. Depth Distribution of ^{60}Co in Concrete Cores Collected Directly Under the Pressure Vessel

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TABLE 4.5. Removal of Radionuclides from Concrete Surface by Stripping Paint Coatings

Concrete Core	Surface Contamination	% Activity Removed by Stripping Paint ^{60}Co	^{137}Cs
PCC-2	Thick Epoxy Coating	99.8	80
PCC-3	Clear Sealer; Smooth Finish	0	*
PCC-6	Thick Epoxy Coating	96	100
PCC-7**	Gray Paint; Smooth Finish	68	0
PCC-9	Chipped Gray Paint; Clear Sealer	13	*
PCC-12	Clear Sealer; Smooth Finish	0	*
PCC-15	Thick Epoxy Coating	46	*
PCC-19	Gray Paint; Smooth Finish	77	*
PCC-20	Gray Paint; Smooth Finish	95	27

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* ^{137}Cs concentrations too low to give an accurate number.

** This core is composed of mildly neutron activated concrete and contains some ^{60}Co incorporated in the concrete with depth.

from the surface to about 2 cm depth indicates the presence of surficially deposited contamination. The ^{60}Co concentration below a depth of approximately 2 cm is derived from neutron activation of trace levels of stable cobalt in the concrete by escaped neutrons from the pressure vessel. There is also the possibility that some of the ^{60}Co at depth results from cross contamination during the coring and/or core segmenting process. Each of the cores show subsurface maxima of ^{60}Co at a depth of about 4 to 6 cm. This is probably due to increased thermalization within the concrete of the neutrons escaping the reactor pressure vessel. The enhanced thermal neutron capture by stable cobalt in the concrete at this depth results in the ^{60}Co maximum. Other neutron activation products detected in these concrete samples were ^{152}Eu and ^{154}Eu (see Appendix C). The core collected from beneath the Turkey Point Unit 4 pressure vessel was obtained within a day of unit shutdown and analyzed as quickly as possible after receipt in the laboratory. As a result, additional activation products resulting from neutron activation of stable elements in the concrete were detected, including ^{46}Sc, ^{59}Fe, ^{51}Cr, ^{141}Ce, ^{65}Zn, ^{58}Co, and ^{124}Sb. The depth distribution of these radionuclides is shown in Figure 4.3. The concentrations have been projected to what would be expected after Turkey Point Unit 4 had operated the equivalent of 30 effective full power years (EFPY). After 30 EFPY the most abundant activation product was ^{55}Fe, whose concentration ($\sim$15,000 pCi/gm) remained fairly uniform to a depth of at least 7 cm. The ^{55}Fe was followed in abundance by ^{152}Eu, ^{60}Co, ^{46}Sc, ^{59}Fe, ^{51}Cr, and ^{154}Eu. Concrete samples collected from the

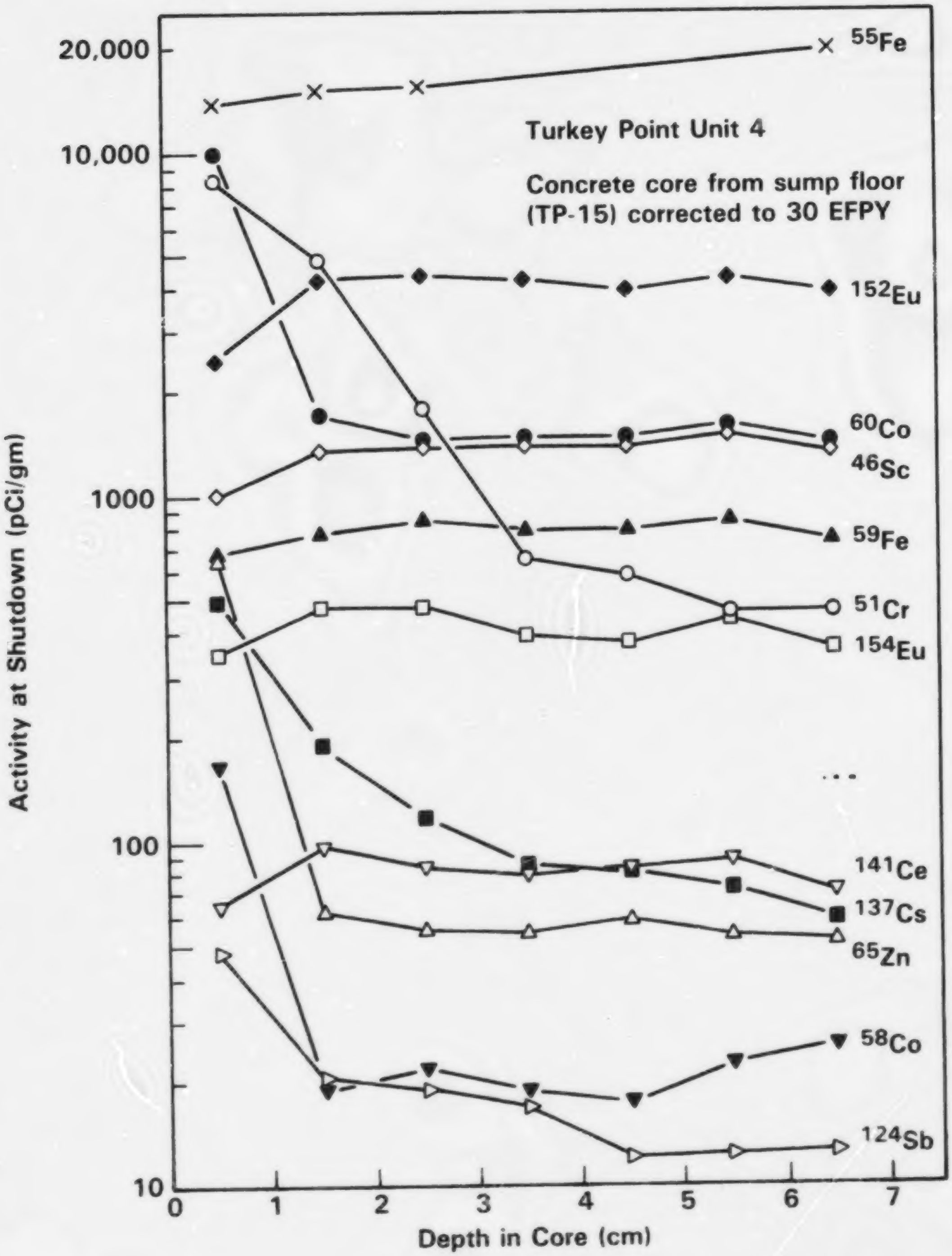

FIGURE 4.3. Depth Distribution of Neutron Activation Products in a Concrete Core Collected Directly Under the Pressure Vessel at Turkey Point Unit 4. Corrected to 30 EFPY.

Humboldt Bay bioshield showed the presence of both surface contamination and in-situ neutron activation products (see Table 4.6). The ^{152}Eu and ^{154}Eu are primarily derived from in situ neutron activation, with the highest concentrations being 309 pCi/gm and 32 pCi/gm, respectively after correction to 30 EFPY.

4.1.5. Contamination Residues in Soils and Sediments

Radionuclide contamination of soils within the exclusion areas of the nuclear power stations was typically limited to small patches of very low concentrations of radionuclides (see Appendix C). Areas included for examination in this program focused upon locations within radiation controlled areas and where spills of radioactive materials (mostly liquids) had occurred. This was a cursory examination of areas of known or suspected contamination and did not seek to address the extent of such contamination. As would be expected from the intent of the sampling, residual radionuclide contamination was usually present; however, the levels of contamination were usually below a suggested residual radionuclide contamination level which would result in an annual dose of 10 mrem/yr.(15) However, several small patches of soils and holding pond sediments at several of the stations contained ^{60}Co and ^{137}Cs concentrations up to several tens to hundreds of pCi/g (on a dry weight basis). Table 4.7 shows the ranges for the detectable gamma emitting radionuclides in exclusion area surface soils, and for ^{238}Pu and $^{239-240}$Pu in Pathfinder and Humboldt Bay exclusion area soils. The most abundant radionuclides were typically ^{60}Co and ^{137}Cs, with occasional trace amounts of other radionuclides including ^{134}Cs, ^{58}Co, ^{54}Mn, ^{106}Ru, ^{110m}Ag, ^{125}Sb, and ^{144}Ce.

4.2. DISTRIBUTIONS AND INVENTORIES OF RESIDUAL CONTAMINATION

A significant part of this research program was the determination of distributions and inventories of residual radionuclide contamination within the various operating systems at the PWR's and BWR's examined. This process entailed utilizing the residual concentrations, as defined by the sampling and analytical phases of the research program, to construct operating system and plant inventories. In order to construct operating system and total plant inventories of residual radionuclides, it was necessary to determine total surface areas present in the various systems upon which the contaminated corrosion films were deposited. The total surface area estimates were constructed by either: 1) using the surface area estimates from the reference PWR or reference BWR studies, corrected by a scaling factor, or, whenever possible, 2) through the use of the specific plant isometrics and prints for the various systems and subcomponents. The radionuclide distributions within the plant systems and the inventories by system are described below, based upon reactor type. The plants were separated by reactor type since there were significant differences between the operating systems of BWR's and PWR's, and thus significant differences between the radionuclide distributions throughout the plant.

TABLE 4.6. Humboldt Bay Concrete Bioshield Samples Activity at Shutdown and 30 EFPY (pCi/gm)

	^{54}Mn	^{55}Fe	^{60}Co	^{63}Cs	^{125}Sb	^{134}Cs	^{137}Cs	^{152}Eu	^{154}Eu	^{155}Eu
HBRS-22 Bottom Inside Edge	42.7		378		0.45	8.2	199	188	11.6	2.2
HBRS-23 Bottom Inside Edge	25.7		821		1.67	30.1	475	309	18.1	3.61
HBRS-24 Top Outside Edge	39.7	7380	303	55	1.41	51.0	544	63	4.9	0.80
HBRS-25	176	16600	1308	41	<0.7	153	72	293	31.6	5.3

TABLE 4.7. Concentration Ranges of Radionuclides in Contaminated Surface Soils (0-4 cm) from Radiation Controlled Areas (pCi/g)

Radionuclide	Pathfinder[1]	Humboldt Bay[2]	Dresden[3]	Monticello[4]	Turkey Point[5]	Rancho Seco[6]
^{54}Mn	<0.005-0.06	0.45-5.5	0.02-0.23	<0.004-<0.02	0.03-0.34	<0.003-0.27
^{60}Co	<0.01-1.7	26-377	1.3-161	0.006-0.45	4.0-45	0.012-11
^{106}Ru	<0.1-0.36	<0.02-<0.05	<0.07-0.2	<0.07	<0.1-0.2	<0.03-<0.09
^{125}Sb	<0.02	<0.4-4.9	<0.03-<1	<0.03	<0.06-22	<0.006-0.75
^{134}Cs	<0.004-<0.01	1.5-6.1	<0.01-6.3	<0.004-0.16	0.28-5.5	0.01-0.95
^{137}Cs	0.15-2.9	25-91	0.49-260	0.068-2.1	1.7-11	0.05-4.9
^{144}Ce	<0.03-	<0.3-1.3	<0.04-1.5	0.083-0.17	<0.05-0.27	<0.02
^{238}Pu	$3\text{-}41 \times 10^{-5}$	$8.2\text{-}170 \times 10^{-3}$	N.M.	N.M.	N.M.	N.M.
$^{239-240}$Pu	$6\text{-}42 \times 10^{-4}$	$9.5\text{-}230 \times 10^{-3}$	N.M.	N.M.	N.M.	N.M.

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1. Fourteen soil samples
2. Five soil samples
3. Four soil samples; highest observed contamination was at a depth of 15-30 cm in D-SS-25B
4. Four soil samples
5. Six soil samples
6. Seven soil samples

N.M. Not measured

4.2.1 Residual Radionuclide Distribution in PWR's

In order to clarify the discussion regarding radionuclide distributions and inventories within PWR's, the operating systems in a PWR, as shown in Figures 4.4 and 4.5, will be described. There are three different coolant loops utilized during power production in a PWR, as shown in Figure 4.4. The first loop is the primary coolant loop. This loop provides heat transfer from the reactor core to the secondary (steam producing) loop. The primary loop is pressurized such that negligible boiling occurs during the heat transfer process. The primary loop gives up its heat at the steam generator. This is the interface between the primary and secondary loops. The secondary loop, or steam loop, provides the steam for the turbine and consequent electrical generation. After passage through the turbine, the secondary steam loop interfaces with the tertiary loop in the condenser. Here the excess heat from the secondary loop is removed and the condensate in the secondary loop proceeds back as feedwater to the steam generators. The heat acquired by the tertiary loop is then rejected to the environment. The advantages of the PWR system for power generation are: 1) additional isolation of radioactive by-products from the environment as a result of the three loops, and 2) slightly increased efficiency of operation since the primary loop can be operated at higher temperatures and pressures than in a boiling water reactor. From a decommissioning standpoint, this type of system should offer more confinement of the radioactive residuals, almost totally within the primary system unless major leakage between the primary and secondary loops occurs. However, the greatly increased surface area of the primary loop contained in the steam generators provides for much greater radioactive corrosion product deposition compared to a BWR.

Figure 4.5 shows a more detailed schematic of the primary reactor coolant system. The figure illustrates a four-loop primary coolant system; the actual number ranges from 2 to 4 for newer PWR's. This figure illustrates the various subcomponents in the primary coolant loop, including the steam generators, the coolant pumps, the pressurizer and pressurizer relief tank, and the coolant piping. Another subcomponent not illustrated in detail is the chemical and volume control system (CVCS) which, as the title indicates, is utilized to provide volume control and maintain proper primary coolant water chemistry. The chemical controls entail the maintenance of appropriate boron concentrations, optimal pH, and removal of nondesirable soluble and particulate species, including activated corrosion products and fission products.

As an initial approximation, one would anticipate that the bulk of the residual radionuclide contamination translocated from the pressure vessel would be present in the steam generators for two reasons. First, the surface area presented by the steam generators represents the overwhelming portion of the total surface area in the primary loop. Secondly, the heat transfer process occurring within the steam generators tends to increase corrosion product deposition processes, thus also tending to increase the residual radionuclide inventory contained on surfaces in the steam generators.

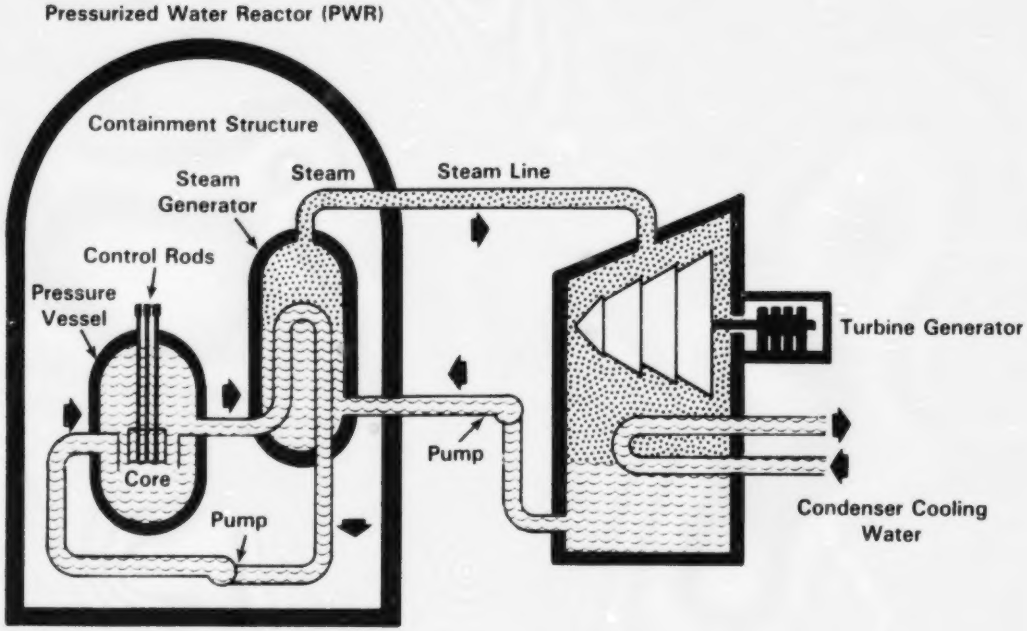
Pressurized Water Reactor (PWR)
Containment Structure
Steam Generator
Steam
Steam Line
Control Rods
Pressure Vessel
Turbine Generator
Core
Pump
Pump
Condenser Cooling Water

FIGURE 4.4. Pressurized Water Reactor (PWR) Schematic Illustrating the Primary, Secondary, and Tertiary Coolant Loops

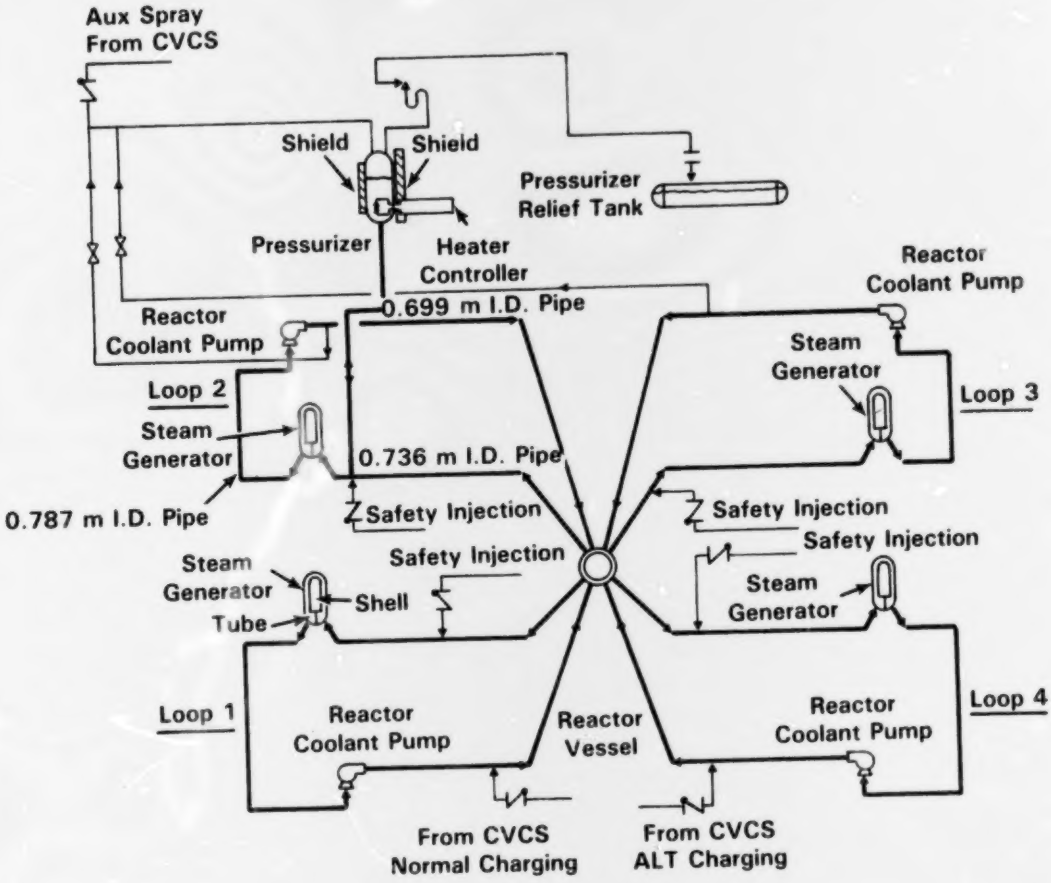
Aux Spray
From CVCS
Shield
Shield
Pressurizer
Relief Tank
Pressurizer
Heater
Controller
Reactor
Coolant Pump
0.699 m I.D. Pipe
Reactor
Coolant Pump
Steam
Generator
Loop 2
Loop 3
Steam
Generator
0.736 m I.D. Pipe
0.787 m I.D. Pipe
Safety Injection
Safety Injection
Safety Injection
Safety Injection
Steam
Generator
Shell
Tube
Steam
Generator
Loop 1
Reactor
Coolant Pump
Reactor
Vessel
Reactor
Coolant Pump
Loop 4
From CVCS
Normal Charging
From CVCS
ALT Charging

FIGURE 4.5. Reactor Coolant System in a Four Loop PWR

The actual distribution of the total residual radionuclide inventory measured throughout the various operating systems for the three PWR's examined in this program is shown in Table 4.8. As expected, the steam generators contained the single largest repository of deposited radionuclides at the PWR stations examined during this program. The percentages of the total residual radionuclide inventories in the steam generators were 77, 89,and 94% for Indian Point One, Turkey Point Unit 3, and Rancho Seco, respectively. The other repository of significance in a PWR is the Radwaste System, which typically contained 5 to 10% of the total residual inventory.

TABLE 4.8. Distribution in Per Cent of the Radionuclide Inventory Estimates for Three Pressurized Water Reactors

	Turkey Point-2	Indian Point-1	Rancho Seco
Steam Generators	89	77	94
Pressurizer	0.5	0.5	0.33
RCS Piping	0.9	2.6	0.71
Piping (Except RCS)	<0.01	14	<0.01
Secondary System	0.1	0.2	0.05
Radwaste	9.2	7	5

Since the steam generator inventory is so important to the overall inventory within the PWR, an additional explanatory section regarding the inventory calculations for the steam generators is included below.

Steam Generator Inventory Estimates. For the Rancho Seco and Turkey Point Stations, radionuclide inventories in the steam generators were calculated using dose rates associated with the steam generator tubes and the relative radionuclide composition as measured in other primary loop samples. Average residual radionuclide concentrations in $\mu Ci/cm^2$ were calculated using: 1) the measured relative radionuclide abundances in primary loop corrosion films, 2) the known value of the dose rate to curie content conversion factor for each major radionuclide contributing to the gamma dose rate, and 3) the gamma dose rate at selected locations in the steam generator. This was accomplished using the ISOSHILD II computer code.(16) For Indian Point One, deposited radionuclide concentrations were based directly upon a scraping sample obtained from the primary side of the No. 14 nuclear boiler. The measured or calculated radionuclide concentrations are shown in Table 4.9 for both ^{58}Co and ^{60}Co at the three reactor stations.

TABLE 4.9. Steam Generator ^{58}Co and ^{60}Co Concentrations Measured or Calculated at Four Reactors

Reactor	^{58}Co μCi/cm^2	^{60}Co μCi/cm^2
Turkey Point 3[A]	4.0	3.0
Turkey Point 4[A]	2.8	2.1
Indian Point 1[B]	-	2.4
Rancho Seco[A]	2.6,3.2	3.5,4.3

[A] Calculated from dose rate conversions
[B] Measured directly from corrosion film scrapings

For Rancho Seco, two values are shown since the dose rates were known in each of the two steam generators. Thus, the values shown represent the surface concentrations present in the two steam generators. These values can be compared to previous data obtained from the primary side of PWR steam generators. We have replotted the data of Bergman, Roesmer, and Perone(17) showing both ^{58}Co and ^{60}Co concentrations versus the number of years of operation, adding the points from our study. The data for ^{58}Co from our study fall well within the range of previous measurements as shown in Figure 4.6. The ^{60}Co data as shown in Figure 4.7 are on the low side of what might be expected for reactor operation periods at the stations examined, but well within the range of an extrapolation of the CORA calculation. The CORA computer program(17) incorporates the more fundamental aspects of transport, activation and other crud-related mechanisms in a nuclear power plant, and predicts several parameters of interest, including radionuclide concentrations in crud layers, crud weight, crud specific activity, and gross radiation levels. This reasonably good agreement lends additional credence to the steam generator inventories presented herein, and is of note since the steam generator inventories are such a significant fraction of the total PWR inventory. As noted earlier, we feel the accuracy of our estimates is within ±50%.

4.2.2. Residual Radionuclide Distribution in BWR's

Four boiling water reactors were examined during this program. These included Pathfinder, Humboldt Bay, Dresden Unit One, and Monticello. The Pathfinder reactor was an early experimental unit having a nuclear superheater. Dresden Unit One was a dual cycle system using both direct primary steam for electrical power generation and also incorporating steam generators for secondary steam production. Humboldt Bay and Monticello are of a conventional BWR design. The power generation system for a typical BWR is shown in Figure 4.8. This system contains only two heat transfer loops, in contrast to the three loops present in a PWR. There is steam production in the primary loop, and this primary steam proceeds directly to

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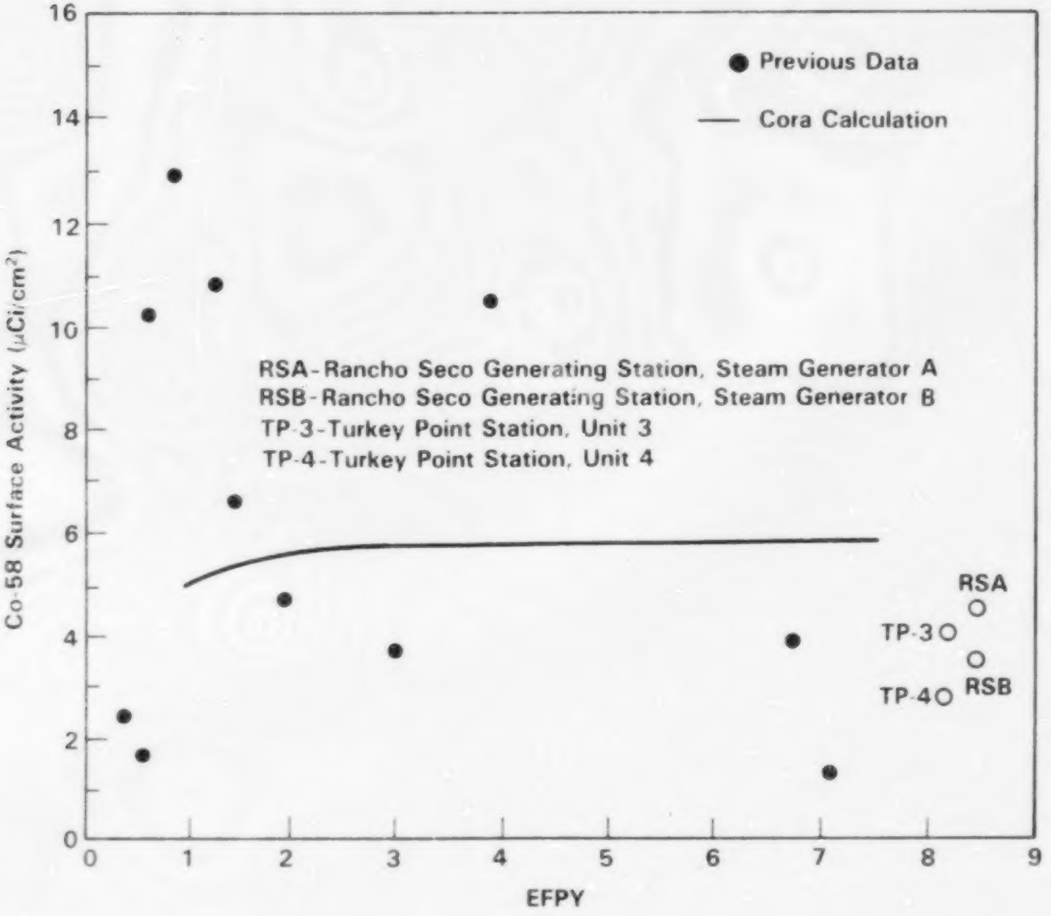


FIGURE 4.6. Comparison of Steam Generator Tube Co-58 Surface Activity Versus EFPY Calculated from this Research with Previous Data (Bergman, Roesmer, and Perone, 1983)

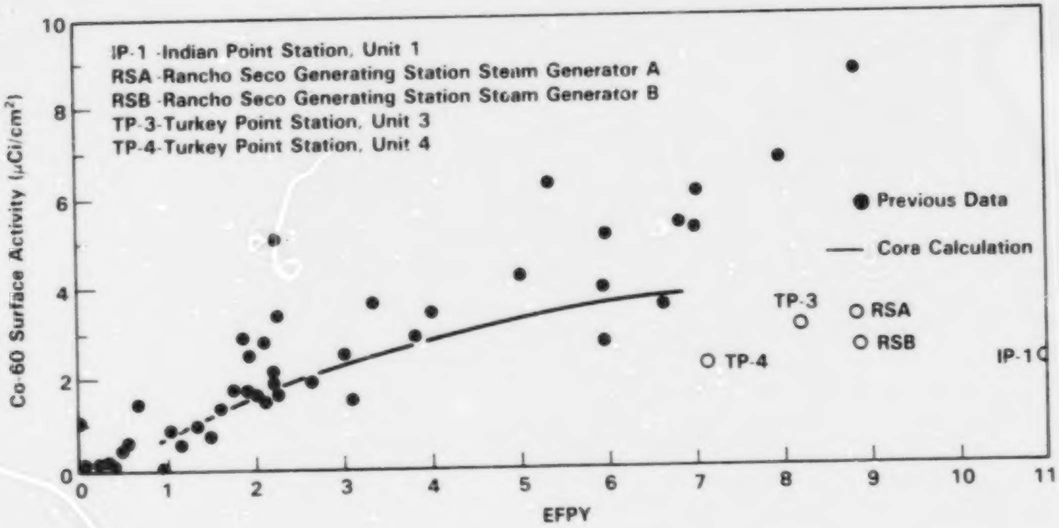
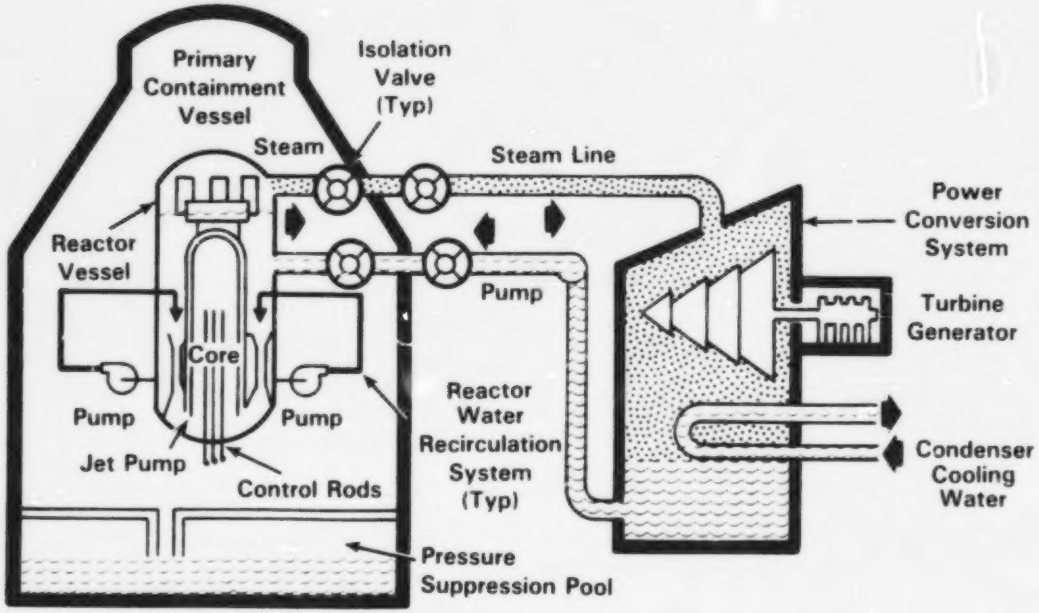


FIGURE 4.7 Comparison of Steam Generator Co-60 Surface Activity Versus EFPY Calculated from this Research with Previous Data (Bergman, Roesmer, and Perone, 1983)



Primary Containment Vessel
Isolation Valve (Typ)
Steam
Steam Line
Power Conversion System
Reactor Vessel
Pump
Turbine Generator
Core
Pump
Pump
Reactor Water Recirculation System (Typ)
Jet Pump
Control Rods
Condenser Cooling Water
Pressure Suppression Pool

FIGURE 4.8. Nuclear Power Generation System for a Boiling Water Reactor (BWR)

the turbine, where electrical generation takes place. The secondary coolant loop removes excess heat from the primary loop in the condenser. From the condenser the primary condensate proceeds into the feedwater stage and the secondary coolant loop removes the excess heat to the environment.

The BWR's operate at lower pressures than PWR's and also gain higher steam temperatures for lower average fuel temperatures than in PWR's. A disadvantage of the BWR is the use of radioactive primary steam, although the most abundant radionuclide is normally the short-lived neutron activation product ^{16}N (7 sec halflife).

One would anticipate that the major residual radionuclide repositories, outside of the reactor vessel for the BWR should be the reactor water cleanup system, the radwaste system, the feedwater system, and perhaps the condenser because of its large surface area.

The residual inventories estimated in the various operating components for the three BWR's sampled in this program (Humboldt Bay, Dresden Unit One, and Monticello) are shown in Table 4.10. As shown in the table, each reactor system has a different residual distribution pattern throughout the various operating systems. Differing reactor designs are responsible for much of the variation observed in the residual radionuclide distribution.

TABLE 4.10. Distributions in Per Cent of the Residual Inventory Estimates in Three Boiling Water Reactors

Operating System	Humboldt Bay	Dresden Unit One	Monticello
Main Steam Lines	0.7	<0.1	1
Condensate	38	17	<0.1
Feedwater System	<0.1	61*	19
Reactor Cleanup System	38	5.2	70
Radwaste System	23	17	10

*Including a secondary steam generation loop which accounts for most of radionuclide inventory in this system.

Humboldt Bay was an early design BWR and used natural feedwater recirculation, thus the feedwater system is a relatively smaller component compared with other BWR's. The large percentage of the total inventory in the condensate system at Humboldt Bay is a result of two factors: 1) the large total surface area associated with this system, and 2) the combination of a less efficient reactor cleanup system (compared to current standards) coupled with less corrosion resistant materials than are presently employed in newer BWR's.

Dresden Unit One also has unique features associated with the design of its operating systems which account for the residual radionuclide distribution observed. Dresden Unit One employed a dual cycle steam system in which primary steam was fed directly to the turbine, and secondary steam was also produced in steam generators. The secondary steam generation component is listed in Table 4.10 as part of the feedwater system, since the primary water exiting the secondary steam generators was recycled to the reactor vessel. The secondary steam generators account for most of the inventory in the feedwater system. The relatively large percentage of the total inventory in the condensate system at Dresden Unit One is probably due to the same reasons noted for Humboldt Bay.

Monticello is the only recent generation BWR examined in this research program. The inventory distribution observed at Monticello is similar to what might be anticipated for a typical BWR. The reactor cleanup system contains by far the largest percentage of the total residual radionuclide inventory (70%). The feedwater system contains the next largest fraction of the total inventory (19%), followed by the radwaste system (10%). The total inventory at Monticello is also considerably lower than observed at the other BWR's when differences in unit sizes are taken into account. In retrospect, it would have been beneficial if another more recent BWR could have been examined during this research program. The additional data would have been valuable, since the inventory at Monticello is so different from that observed at Humboldt Bay and Dresden Unit One.

4.2.3 Radionuclide Inventories in the Nuclear Power Plants

Table 4.11 contains the total estimated inventories for the seven sites examined, as well as the electrical ratings, and the approximate number of operational years for the units at the time of our inventory construction.

Except for Pathfinder, the operational periods ranged from 8 years for Turkey Point Unit 3 to slightly over 18 years for Dresden Unit One. As a first approximation, one would expect that the residual radionuclide inventories would be related to unit size and length of service and that a generic relationship could be found which would be of use for predicting inventories at other nuclear power stations. In Table 4.12 we have listed the two simplest relationships and applied them for all seven sites and again for five of the sites (excluding Pathfinder and Monticello which appear anomalous).

TABLE 4.11. Comparison of Total Residual Radionuclide Inventories and Operating Parameters for the Seven Nuclear Generating Stations Examined

Stations	Total Inventory* (Curies)	Years of Operation	MWe	Reactor Type
Pathfinder	57	3.5	58	BWR
Humboldt Bay	600	13	63	BWR
Dresden-1	2350	18.3	210	BWR
Monticello	514	10	550	BWR
Indian Point-1	1050	11	170	PWR
Turkey Point-3	2580	8.3	660	PWR
Rancho Seco	4470	8.8	935	PWR

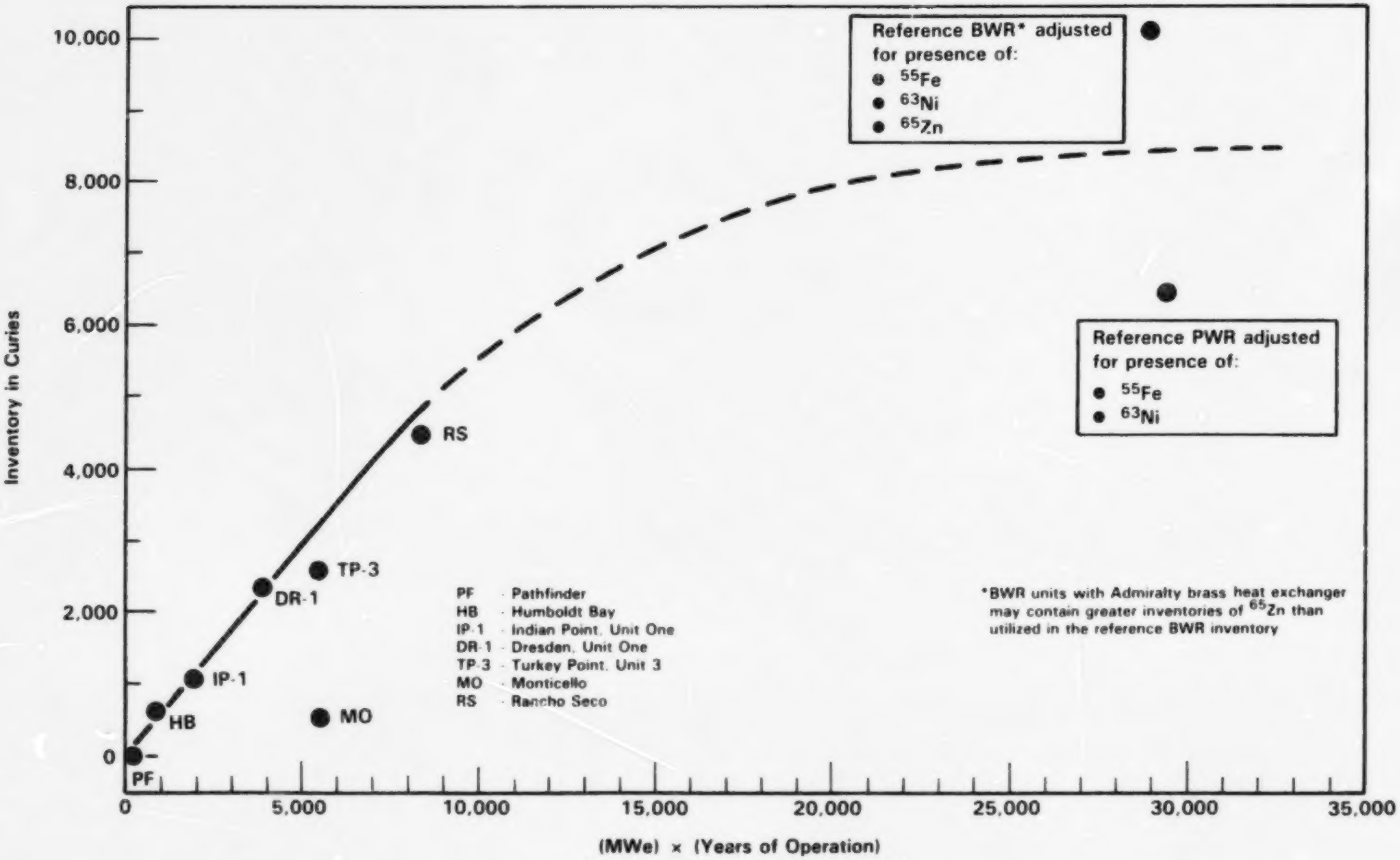
*Inventory includes radionuclides with half-lives greater than 245 days (^{65}Zn).

TABLE 4.12. Relationship of Total Inventory at the Nuclear Generating Plant to Selected Operating Parameters.

Relationship	Curies Mean and Std. Dev. (7 Sites)	Curies Mean and Std. Dev. (5 Sites)*
Inventory/MWe	7.12 ± 3.12	5.2 ± 4.2
Inventory/(MWe x Yrs operation)	0.43 ± 0.28	0.58 ± 0.10

*Excluding Pathfinder and Monticello

The first relationship normalizes the total radionuclide inventory strictly to the unit size in MWe. The relationship appears more applicable for the seven sites than when excluding Monticello and Pathfinder, as evidenced by the lower standard deviation about the mean. The second potential correlation is a nearly linear relationship developed when the total radionuclide inventory is normalized by the product of unit size (MWe) and length of service (years). This is also shown graphically in Figure 4.9. For the seven site comparison, this relationship is not as good as that developed when normalizing the inventory to just the unit



*BWR units with Admiralty brass heat exchanger may contain greater inventories of ^{65}Zn than utilized in the reference BWR inventory

FIGURE 4.9. Relationship of Total Inventories Versus Unit Size and Years of Operation for Seven Reactors and the Reference BWR Oak et al, 1980) and Reference PWR (Smith et al, 1978)

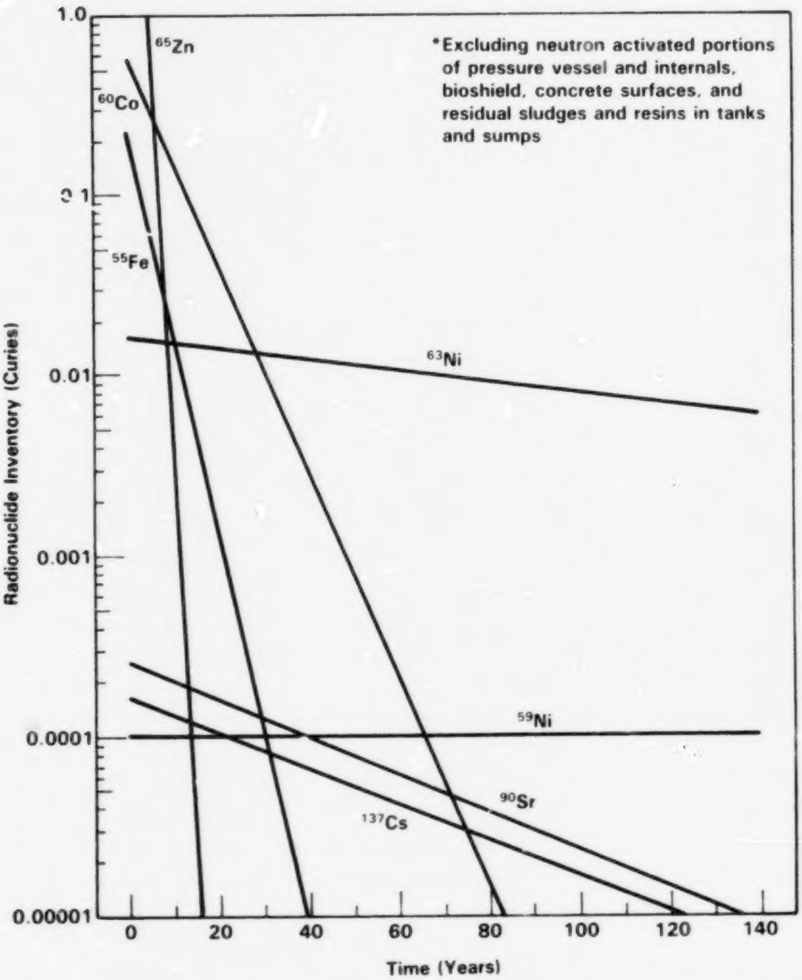
size. However, when Pathfinder and Monticello are deleted, the relationship improves greatly, having a standard deviation about the mean of only 17%. The exclusion of Pathfinder can be argued justifiably, since it was a unique, early design reactor, with a very short operational period not extending beyond testing phases. Monticello, on the other hand, is a large, relatively new BWR. It has been excluded in the five site case because the data are somewhat anomalous, both in total inventory and radionuclide composition. It may be, however, that Monticello is typical of newer BWR's, at least to the degree that they will have lower residual inventories. An argument can be made for this point based upon yearly radwaste shipments.(18) BWR's, on the average, ship offsite some 5 to 6 times as much radwaste as do PWR's in the form of spent resin wastes. Thus PWR's may be accumulating relatively more waste over an extended period of time, whereas BWR's are shipping this inventory offsite in the form of rad-waste. Also, BWR's undergo more frequent primary system chemical decontamination operations relative to PWR's, which would further reduce the residual radionuclide inventories.

4.2.4. Changes in Radionuclide Inventory with Time after Shutdown

Since the residual radionuclide contamination is a mixture with varying half-lives, the relative composition of the radionuclides present will change, along with the decrease in absolute concentrations, as radioactive decay occurs with time after shutdown. Initially, significant radionuclides, e.g. ^{60}Co, ^{55}Fe, ^{58}Co, ^{65}Zn, will decay rapidly in comparison to radionuclides with longer half lives, e.g. ^{59}Ni, ^{63}Ni, ^{137}Cs.

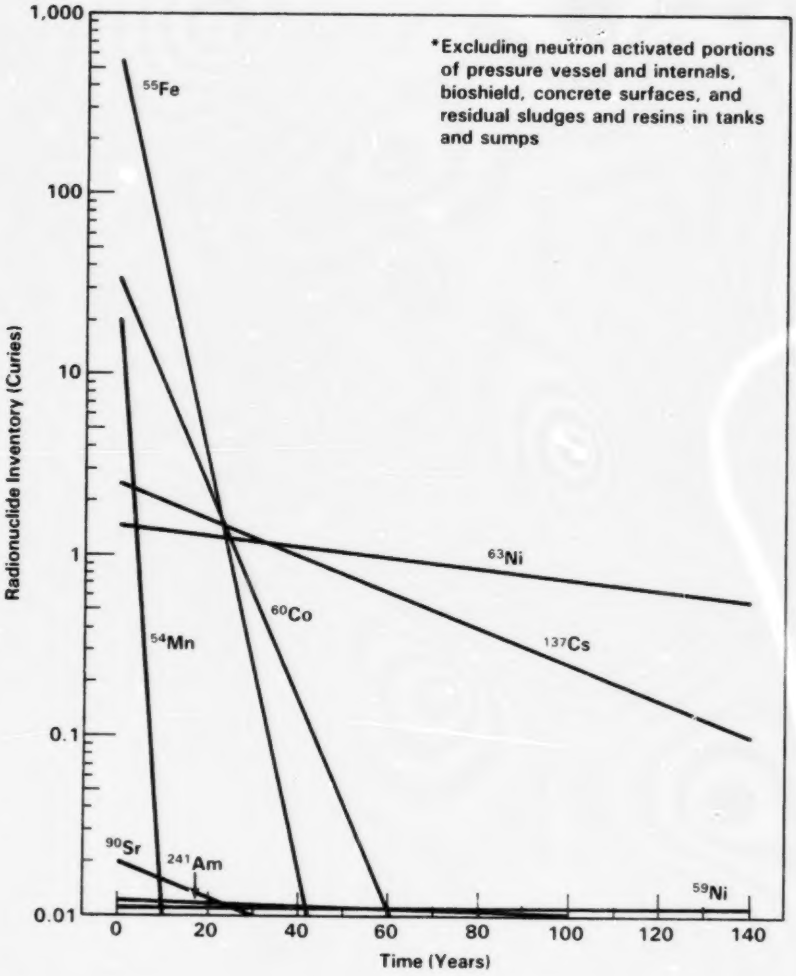
The relative abundances of radionuclides versus time after shutdown at the seven reactor sites examined in this research program are shown in Figures 4.10 through 4.16. It should be noted that these decay plots are for the total estimated residual radionuclide inventory after shutdown, and as such they reflect to a great degree the primary coolant system, since the majority of the inventory is present in this system. Other systems may show somewhat differing behavior, especially the radwaste and fuel pool areas, since the ^{134}Cs and ^{137}Cs are typically enhanced in these systems.

Several important conclusions can be reached from examination of the seven decay plots. The range in inventories and concentrations of the individual radionuclides of concern covers many orders of magnitude at a given station. Iron-55, ^{60}Co, and at some BWR stations, ^{65}Zn are typically the most abundant radionuclides present shortly after shutdown. The concentrations of these initially abundant radionuclides decreases 3 to 5 orders of magnitude, typically from thousands or hundreds of curies to tens of curies or less, in the first fifty years after shutdown. Although the ^{55}Fe concentrations are relatively high, they are practically inconsequential from a decommissioning standpoint, since the ^{55}Fe concentrations associated with contaminated piping and hardware would always be considerably below the 10 CFR 61 Class "A" waste limit; and since ^{55}Fe emits only a 5.9 keV X-ray, it would contribute negligibly to the external whole-body gamma dose rate. The major contributor to the



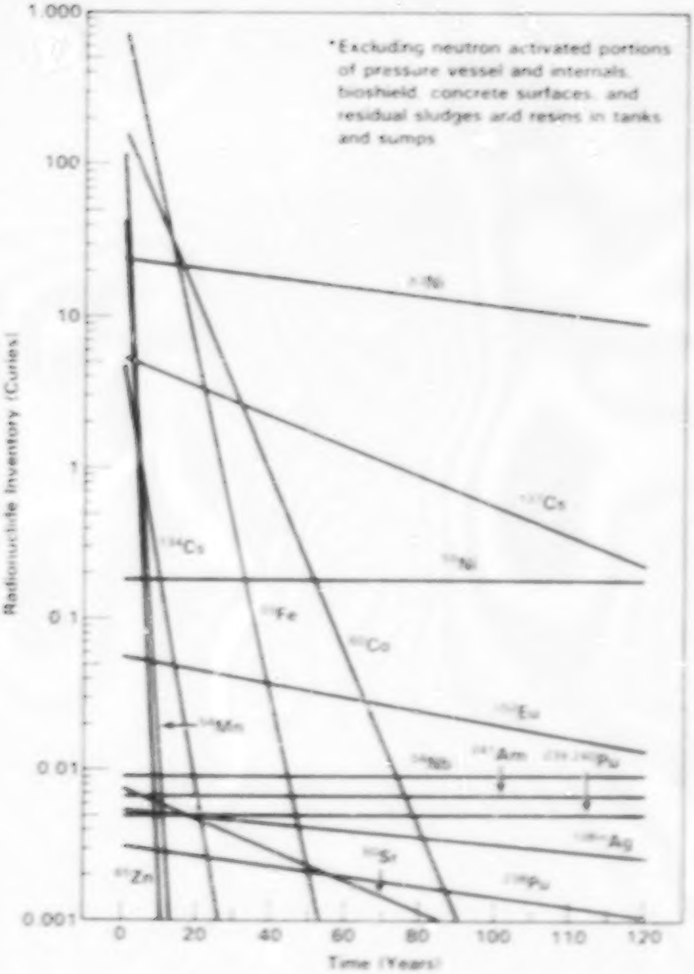
*Excluding neutron activated portions of pressure vessel and internals, bioshield, concrete surfaces, and residual sludges and resins in tanks and sumps

FIGURE 4.10. Pathfinder Nuclear Generating Station. Radionuclide Inventory* Versus Time After Shutdown



*Excluding neutron activated portions of pressure vessel and internals, bioshield, concrete surfaces, and residual sludges and resins in tanks and sumps

FIGURE 4.11. Humboldt Bay Nuclear Generating Unit. Radionuclide Inventory* Versus Time After Shutdown



*Excluding neutron activated portions of pressure vessel and internals, bioshield, concrete surfaces, and residual sludges and resins in tanks and sumps

FIGURE 4.12. Indian Point Station, Unit One. Radionuclide Inventory* Versus Time After Shutdown

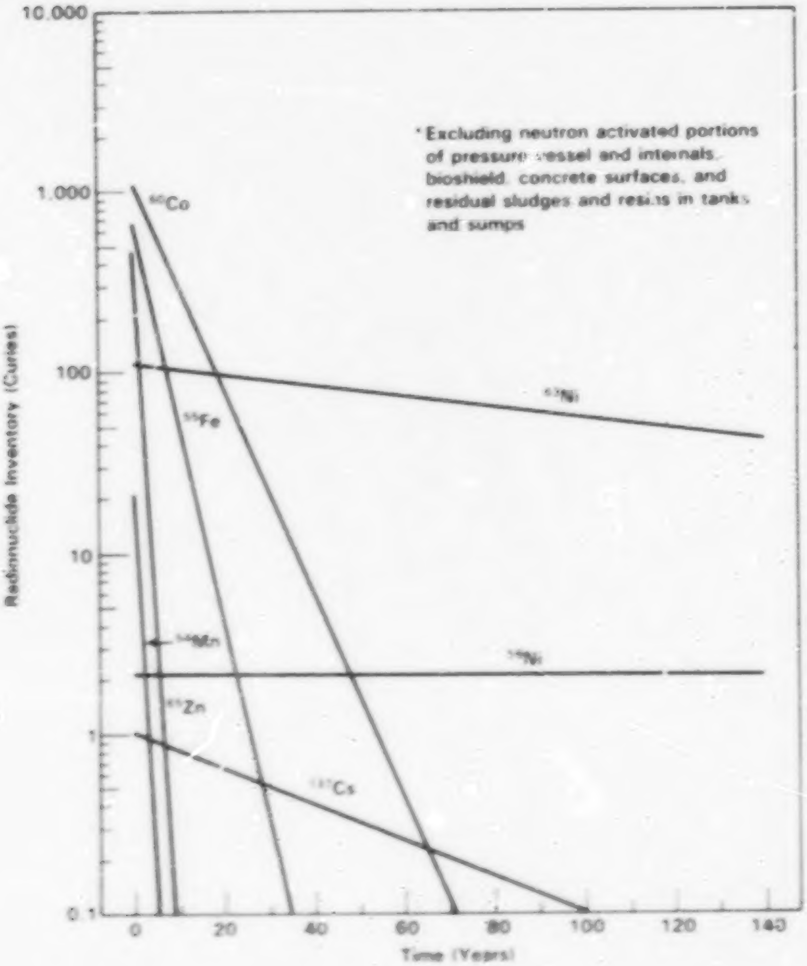
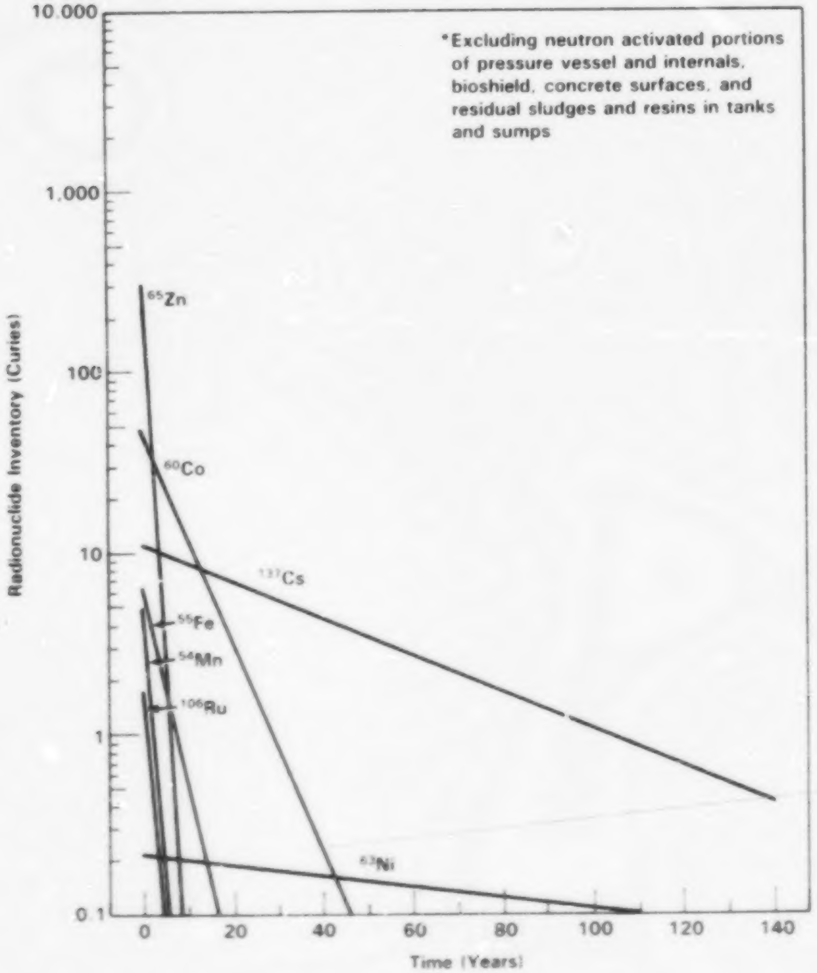


FIGURE 4.13. Dresden Nuclear Station, Unit One. Radionuclide Inventory* Versus Time After Shutdown



*Excluding neutron activated portions of pressure vessel and internals, bioshield, concrete surfaces, and residual sludges and resins in tanks and sumps

FIGURE 4.14. Monticello Nuclear Generating Station. Radionuclide Inventory* Versus Time After Sampling, Assuming Unit Shutdown

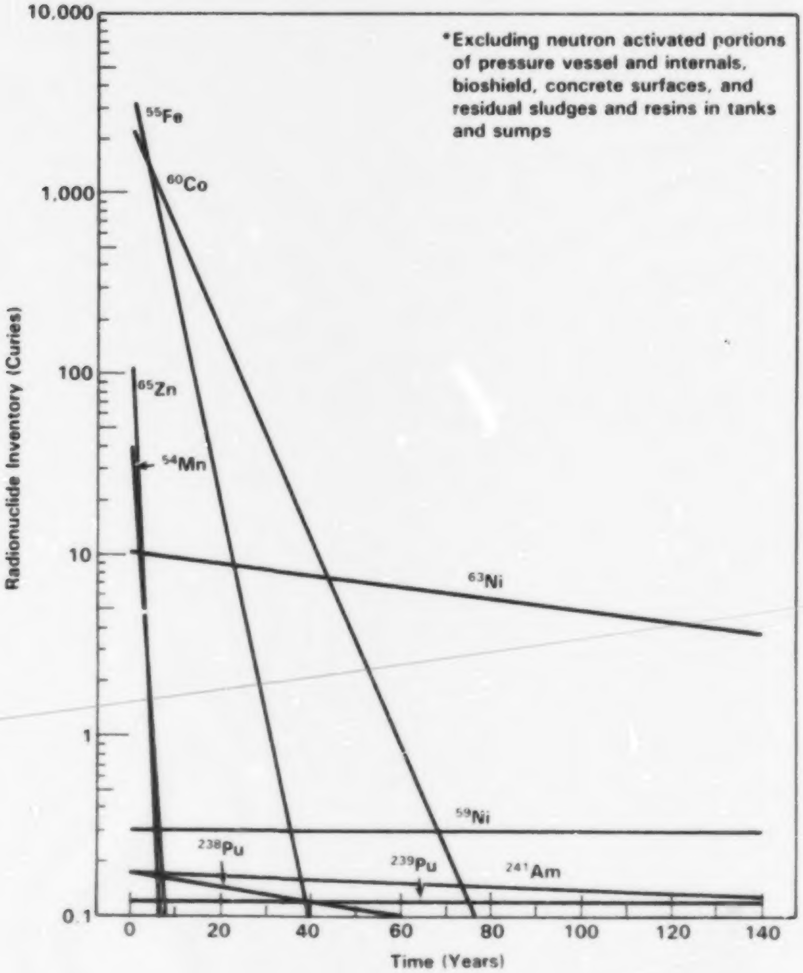


FIGURE 4.15. Turkey Point Station, Unit Three. Radionuclide Inventory* Versus Time After Sampling, Assuming Unit Shutdown

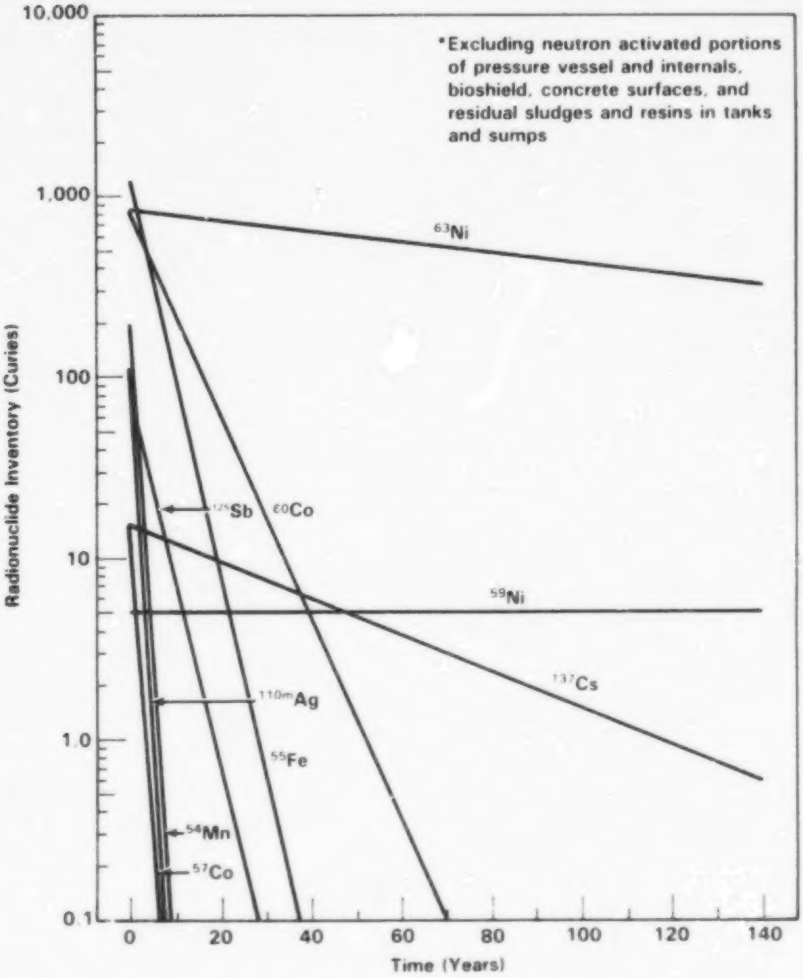


FIGURE 4.16. Rancho Seco Nuclear Generating Station. Radionuclide Inventory* Versus Time After Sampling, Assuming Unit Shutdown

external whole-body gamma dose rate from residual contamination for the first 20 to 60 years at most stations would be ^{60}Co. Zinc-65 would be the major contribution to the whole body gamma dose rate at Pathfinder and Monticello for the first two or three years after shutdown. After 20 to 60 years time the major dose rate contributor would generally become ^{137}Cs. Nickel-63 would become the most abundant radionuclide in all cases except Monticello after a period of 20 to 40 years. However, like ^{55}Fe, the ^{63}Ni concentrations would not be a significant radiological factor affecting decommissioning since: (1) it's concentrations would always be below Class "A" disposal limits, and (2) it decays by emission of only 67 keV (max.) beta particles which would not contribute significantly to the gamma dose rate.

Only after hundreds of years would the transuranic radionuclides become the most abundant radionuclides. However, their concentrations would be extremely low, and always below Class "A" waste disposal limits.

4.3 RELATIONSHIP OF SOURCE MATERIALS AND RESIDUAL NEUTRON ACTIVATION PRODUCTS

The primary factors influencing both the magnitude of the total residual radionuclide inventory and the relative radionuclide abundances within the inventory are: 1) the elemental composition and purity of materials used in construction of the reactor systems; 2) the general design of the primary and secondary systems; 3) the operational parameters including water chemistry, corrosion control, and radwaste management; 4) the fuel integrity; 5) the reactor power level; and 6) the length of operation.

The significant long-lived residual radionuclides produced by neutron activation of source materials at the seven stations examined were ^{55}Fe, ^{60}Co, ^{59}Ni, ^{63}Ni, ^{65}Zn, and ^{110m}Ag. Although ^{94}Nb has been shown to be produced in relatively large amounts in neutron activated stainless steel components inside the reactor pressure vessel, it has never been a significant radionuclide in the primary loop-corrosion product deposits. The important source materials for each, and the impact that source materials, unit design, and operational history have had on overall inventories for the seven units examined in this study are discussed below.

4.3.1. Iron-55 Production

This radionuclide is produced by neutron activation of the 5.8 percent abundant stable iron isotope, ^{54}Fe. After production by neutron activation in the core, translocation of this and other radionuclides from the reactor vessel through the coolant systems will be a function of corrosion and deposition rates. Since carbon steels are more susceptible to corrosion than stainless steels or nickel alloys such as inconel under typical reactor conditions, ^{55}Fe should be more abundant in the translocated inventory in reactors employing larger relative amounts of carbon steel in the primary coolant loop. BWR units employ larger relative amounts of carbon steel than do PWR's, and thus typically contain more ^{55}Fe in the corrosion films. The abundance of translocated ^{55}Fe will also be affected by the chemical

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controls maintained in the coolant loop, e.g., pH and oxygen levels, since these influence corrosion rates. Finally, the type and effectiveness of the reactor water cleanup system will also be a determining factor in the residual ^{55}Fe abundances.

The large fraction of ^{55}Fe in the Humboldt Bay inventory can be explained by the following factors. First, it was an early BWR and made extensive use of carbon steel. Second, during the early history of reactor operations, optimal corrosion controls were not as well developed. Third, the reactor water cleanup system at Humboldt Bay was not as effective as those in newer reactors since the efficiency of reactor water cleanup systems has also improved with time and experience. Finally, Humboldt Bay used stainless steel clad fuel during earlier operations, thus increasing the iron inventory directly exposed to the most intense neutron flux. The large ^{55}Fe inventory at Indian Point Unit One reflects some of the same factors as noted for Humboldt Bay.

The relatively small inventory of ^{55}Fe at Dresden One may be a reflection of its unique design, and the use of materials such as admiralty brass, muntz, and monel in the primary loop which resulted in higher relative abundances of other radionuclides. Also, the fuel elements as well as parts of the fuel support structures were clad with zircaloy-2. Dresden One was a dual cycle reactor system which contained components typical of a BWR and a PWR. Thus, there was more stainless steel present than in other BWR's of the same vintage. This would also tend to increase the ^{60}Co in relation to ^{55}Fe. Dresden One also had admiralty brass (Cu-Zn), monel (Ni-Cu-Co), muntz (Cu-Zn), and copper-nickel alloys present in the reactor system. These metals are somewhat more prone to corrosion. Corrosion of these alloys and translocation of the corrosion products throughout the reactor operating systems, coupled with neutron activation of corrosion products transported into the core region, resulted in higher relative abundances of radionuclides other than ^{55}Fe in the residual inventory.

The ^{55}Fe abundance at Monticello was very low, which may be reflective of several factors. It is a newer BWR employing more modern corrosion controls, zircaloy-2 clad fuel, and a reactor water cleanup system reflective of improved technology. The very low total radionuclide inventory at Monticello was also a reflection of strict corrosion control, well controlled water chemistry, and good radwaste management practices.

The abundances of ^{55}Fe at Turkey Point and Rancho Seco could be representative of the levels in large, modern PWR's. The ^{55}Fe relative abundances at these stations were both close to the mean ^{55}Fe abundance, 44%, calculated for all seven stations.

4.3.2. Cobalt-60 Production

Cobalt-60 is produced by neutron activation of the 100% abundant stable cobalt isotope, ^{59}Co. Cobalt is a trace constituent in both carbon and stainless steels (ranging from 80 to 150, and 230 to 2600 ppm, respectively)(13). Cobalt is also present in inconel and monel. After production in the reactor core, translocation and deposition of ^{60}Co throughout the operating systems will be a function of: 1) corrosion controls, 2) the effectiveness of the reactor water cleanup system, and 3) radwaste management practices.

The inventory of ^{60}Co at Humboldt Bay was rather low, reflecting the primary use of carbon steel, which contains lower cobalt impurity levels than stainless steel, throughout the operating primary systems. The ^{60}Co inventory at Dresden One was significantly higher than at Humboldt Bay for several reasons. First, Dresden One employed monel heat exchanger tubes, which had a cobalt concentration of 5%, in the last three stages of the primary and secondary feedwater heaters. Second, the Dresden One unit was a dual cycle system and incorporated more stainless steel than at Humboldt Bay. Finally, the longer operational history of Dresden One should have resulted in a higher relative abundance of ^{60}Co compared to ^{55}Fe and other shorter-lived radionuclides, due to saturation effects.

The ^{60}Co inventory at Monticello undoubtedly reflected the increased use of stainless steel in newer BWR units, while the lower relative inventory of ^{60}Co at Indian Point One reflected the elevated concentrations of other trace metals introduced from the carbon steel and admiralty brass components of the secondary system.

The larger relative abundance of ^{60}Co at Turkey Point reflected the larger proportion of stainless steel in more recent PWR systems, while the relatively low inventory at Rancho Seco reflected the extensive use of inconel, which contains a lower cobalt concentration than stainless steel.

4.3.3. Zinc-65 Production

This radionuclide is produced by neutron activation of the 49% abundant stable isotope, ^{64}Zn, and appeared in significant concentrations in early BWR units which employed heat exchangers containing admiralty (29% Zn) or muntz metal (40% Zn). These are exemplified by Pathfinder, Humboldt Bay, Monticello, and Dresden in the seven reactors examined in this program. The use of an admiralty brass heat exchanger at Indian Point One is apparent in the residual inventory, even though this exchanger was employed in the secondary system. The condensed secondary steam, which was contaminated in the condenser with stable zinc, was used for primary water makeup, thus bringing this zinc contamination into the primary reactor loop where the stable zinc became neutron activated.

4.3.4. Nickel-59 and ^{63}Ni Production

These two radionuclides are produced by neutron activation of the stable isotopes ^{58}Ni and ^{62}Ni, which have stable isotopic abundances of 68% and 4%, respectively. Nickel alloys, monel and copper-nickel, were used in heat exchangers in some early reactors. More recently, inconel (60-80% nickel) has received extensive use in reactor systems, both for reactor internals and heat exchanger surfaces. The inventory at Dresden One reflected the latter application, with the use of both monel (67% nickel) and copper-nickel (30% nickel) heat exchanger surfaces in the reactor feedwater heating system.

The relatively large ^{63}Ni inventory at Rancho Seco reflected the more recent trend of extensive use of inconel for the reactor internals and also in the steam generators as a heat exchange surface. At Rancho Seco this extensive use has resulted in ^{63}Ni becoming the third most abundant radionuclide after ^{55}Fe and ^{60}Co. Nickel-63 constituted 25% of the total inventory at Rancho Seco.

4.3.5. Silver-110m Production

The only reactor unit in which ^{110m}Ag was a significant contributor to the overall inventory was Rancho Seco. This radionuclide is produced by neutron activation of the 48% abundant stable isotope, ^{109}Ag. The large relative abundance of ^{110m}Ag in the Rancho Seco inventory is explained by the presence of silver-indium-cadmium control rods. Like other reactor components, the control rods are subject to corrosion and erosion processes which can lead to contamination of the primary coolant. Traces of ^{110m}Ag were also observed at Monticello associated with a silver alloy gasket used for sealing the head of the reactor pressure vessel. Silver-108m ($t_{1/2}$ = 130 y) is also produced by thermal neutron capture on 52% abundant ^{107}Ag. However, its low cross section (0.33 barns) and long half-life limit its production. Nevertheless, the use of large amounts of silver in PWR control rods will result in a large inventory of ^{108m}Ag in these components.

4.3.6. Cobalt-58 and ^{54}Mn Production

Cobalt-58 and ^{54}Mn are both produced by fast neutron reactions with parent elements contained in the reactor core structural materials. The ^{58}Co is primarily produced by the ^{58}Ni (n,p) ^{58}Co reaction, which has an average cross section in a fission neutron spectrum of 100 mb. The target isotope, 67.8% abundant ^{58}Ni, is present in the inconel (60-80% Ni) and stainless steel (8-12% Ni) used in the pressure vessel and primary loop. Although the half-life of ^{58}Co is short (71 days), it is one of the most abundant radionuclides in corrosion product residues immediately after reactor shutdown.

The ^{54}Mn is produced primarily by the ^{54}Fe (n,p) ^{54}Mn reaction, which has an average cross section in a fission neutron spectrum of 53 mb. The target isotope, 5.84% abundant ^{54}Fe, is present in the steel construction materials of the pressure vessel, fuel support structures, and the primary loop. Corrosion of the steel components can transport iron to the fast neutron flux region of the reactor where the ^{54}Mn is produced.

4.3.7. Niobium-94 Production

Niobium-94 ($t_{1/2}$ = 20,000 y) is produced by thermal neutron capture from the 100% abundant stable isotope, ^{93}Nb, which has a cross section of 1.15 barns. It decays by beta emission with a maximum beta energy of 473 keV to a single level of ^{94}Mo at 1574 keV. A cascade of two 100% abundant gammas of 703 and 871 keV each results. The presence of relatively high levels of niobium in stainless steel (5-300 ppm) and inconel (390-50,000 ppm) would lead to production of significant amounts of the very long-lived ^{94}Nb in reactor core materials.(13) For long deferral intervals prior to decommissioning, ^{94}Nb may in fact represent the principal contributor to personnel exposure during dismantlement of the reactor pressure vessel. However, the extreme insolubility of niobium does not permit significant translocation from the pressure vessel and deposition in other plant systems. Therefore, ^{94}Nb has been a very minor constituent of the residual radionuclide deposits in plant systems.

5.0 RESIDUAL RADIONUCLIDE CORRELATIONS FOR 10 CFR 61 WASTE CLASSIFICATION

Radionuclide correlations are presently being utilized by the utilities to estimate the concentrations of a number of 10 CFR 61 radionuclides in typical waste streams (e.g., resins, sludge, DAW, etc.) being prepared for disposal. This is usually accomplished by performing comprehensive sampling and radiochemical analyses of these waste forms on enough samples to provide reasonably valid correlation factors for those radionuclides not measurable by gamma-ray spectrometry. Once adequate correlation factors have been established relative to some easily measurable gamma emitting radionuclide (e.g. ^{60}Co, ^{137}Cs, ^{144}Ce), the waste stream samples can be analyzed by gamma-ray spectrometry, and the non-gamma emitting radionuclide concentrations can be estimated from the established correlations.

However, the typical waste streams of resin, sludge, evaporator bottoms, or DAW do not have the same relative radionuclide concentrations as the residual radionuclide contamination deposited on piping and hardware. Therefore, the radionuclide concentrations in the most representative piping samples which had been exposed to primary coolant at six of the nuclear plants examined in this study (Pathfinder was excluded as being atypical) were evaluated to determine if adequate, generic correlation factors could be established for these types of samples. Table 5.1 lists the correlation factors developed when normalizing the concentrations of 10 CFR 61 radionuclides to ^{60}Co, which is generally the most readily measured gamma emitter in contaminated primary loop piping and hardware. As shown in the table, the correlations were highly variable. The best correlation observed was the $^{239-240}$Pu/^{60}Co ratio, which gave an average value and associated standard deviation (1σ) of $8.3 \pm 6.8 \times 10^{-5}$ for the six contaminated piping samples. This correlation factor could prove useful for estimating the concentrations of $^{239-240}$Pu relative to ^{60}Co. The average ^{241}Am/^{60}Co and ^{244}Cm/^{60}Co ratios were $15 \pm 15 \times 10^{-5}$ and $9.0 \pm 9.3 \times 10^{-5}$, respectively. Thus, the transuranic radionuclide concentrations could be estimated to within about an order of magnitude from the ^{60}Co concentrations.

The only other potentially useful correlations relative to ^{60}Co were for the ^{55}Fe/^{60}Co and ^{63}Ni/^{60}Co ratios. If the atypically low ^{55}Fe/^{60}Co value for the Monticello sample, M-101, is excluded, the five other samples give an average ^{55}Fe/^{60}Co ratio of 3.8 ± 2.7. This would allow the ^{55}Fe to be estimated within a factor of about five to ten from the ^{60}Co measurement. If the atypically high ^{63}Ni/^{60}Co ratio for the Rancho Seco sample, RSH-2, is excluded, the five other samples give an average ^{63}Ni/^{60}Co ratio of 0.060 ± 0.063.

Cesium-137 and ^{99}Tc do not correlate well with ^{60}Co, and ^{90}Sr correlations are only slightly better. Not enough positive values for the ^{94}Nb/^{60}Co and ^{129}I/^{60}Co ratios were available to determine if adequate correlations existed between these radionuclides, but a range of less-than ratios could prove useful for estimating less-than values of ^{94}Nb and ^{129}I in individual nuclear plant piping and hardware samples.

TABLE 5.1. Activity Ratios Normalized to ^{60}Co in Piping Exposed to Primary Coolant (Decay Corrected to Shutdown Date or Sampling Date)

Sample	^{137}Cs/^{60}Co (x 10^{-3})	^{90}Sr/^{60}Co (x 10^{-4})	^{94}Nb/^{60}Co (x 10^{-4})	^{55}Fe/^{60}Co	^{63}Ni/^{60}Co	^{99}Tc/^{60}Co (x 10^{-5})	^{129}I/^{60}Co (x 10^{-5})	$^{239-240}$Pu/^{60}Co (x 10^{-5})	^{241}Am/^{60}Co (x 10^{-5})	^{244}Cm/^{60}Co (x 10^{-5})
IPH-22$^{(A)}$	11	0.50	6.1	4.8	0.15	0.62	19	3.3	4.4	0.53
HBRS-26$^{(B)}$	1.0	4.3	<4	6.7	0.043	2.5	<7	20.0	37.0	13.0
RSH-2$^{(C)}$	13	<20	<2	5.6	3.42	<2	<1	5.5	4.4	3.8
M-101$^{(D)}$	249	20	<10	0.056	0.0046	3.3	<10	13.0	30.0	26.0
TP-H7$^{(E)}$	0.17	0.37	<0.9	1.3	0.0044	27	<8	5.5	7.5	4.4
D-SC-18$^{(F)}$	0.45	0.53	<0.4	0.61	0.098	0.034	<0.8	2.7	4.1	6.3
AVERAGE ± 1σ	---	---	---	3.8±2.7*	0.060±0.063**	---	---	8.3±6.8	15±15	9.0±9.3

A Corrosion film scrapings from Indian Point-1 steam generator
B Regenerative heat exchanger piping from Humboldt Bay
C Corrosion film deposits on HEPA filter used in steam generator repair
D Reactor water cleanup piping from Monticello
E Alumina grit used for decontamination of Turkey Point Unit 3 steam generator
F Reactor steam vent piping from Dresden-1

* Excludes atypically low value for sample M-101
** Excludes atypically high value for sample RSH-2

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Table 5.2 lists the correlation factors for 10 CFR 61 radionuclides for these samples relative to the gamma-emitting fission product ^{137}Cs. Since ^{137}Cs is more soluble in the primary coolant loop and does not follow the usual corrosion product deposit processes, it's correlations with the concentrations of the transition metal activation products and TRU radionuclides were poor. Not unexpectedly, the ^{90}Sr provided the best correlation with the ^{137}Cs, and gave an average ratio ($\pm 1\sigma$) for the six samples of 0.16 ± 0.16 (Sr/Cs). Neither Cs nor Sr are efficiently sorbed into corrosion films. This ratio indicates that Sr would tend to be more soluble since Sr and Cs are produced in nearly equal abundance during fission.

Table 5.3 lists the correlation factors calculated relative to $^{239-240}$Pu. As expected, the correlations for the other TRU radionuclides, ^{241}Am and ^{244}Cm, were the least variable, giving average ^{241}Am/$^{239-240}$Pu and ^{244}Cm/$^{239-240}$Pu ratios of 1.5 ± 0.5 and 1.1 ± 0.8, respectively. The ^{55}Fe/$^{239-240}$Pu ratio gave an average value of $0.66 \pm 0.57 \times 10^5$, if the atypical Monticello sample, M-101, was excluded.

In summary, generic radionuclide correlation factors for estimating the concentrations of residual radionuclides in contaminated piping and hardware have not proven very consistent. The Pu/^{60}Co, ^{55}Fe/^{60}Co, ^{63}Ni/^{60}Co and ^{90}Sr/^{137}Cs relationships may prove to be most useful. During decommissioning it is obvious that rather comprehensive sampling and radiochemical analyses of 10 CFR 61 radionuclides will need to be performed at each retired nuclear power station to establish residual radionuclide concentrations and associated waste classifications for the various wastes which are generated.

TABLE 5.2. Activity Ratios Normalized to ^{137}Cs in Piping Exposed to Primary Coolant (decay corrected to shutdown date or sampling date)

	$^{60}Co/^{137}Cs$	$^{90}Sr/^{137}Cs$	$^{94}Nb/^{137}Cs$	$^{55}Fe/^{137}Cs$	$^{63}Ni/^{137}Cs$	$^{99}Tc/^{137}Cs$	$^{129}I/^{137}Cs$ $(\times 10^{-6})$	$^{239-240}Pu/^{137}Cs$	$^{241}Am/^{137}Cs$	$^{244}Cm/^{137}Cs$
IPH-22[a]	61	0.0045	0.051	436	34	0.00054	1.7	0.0030	0.0085	0.00048
HBR-26[b]	100	0.41	<0.4	6700	43	0.025	<7	0.26	0.17	0.33
RSH-2[c]	77	<0.15	<0.015	430	760	0.0015	<0.08	0.0047	0.0034	0.0029
M-101[d]	4	0.0080	<0.004	0.22	0.018	0.00013	0.04	0.00052	0.0017	0.00033
TP-AG[e]	5900	0.22	<0.5	7650	26	1.6	<0.005	0.32	0.44	0.26
DSC-38[f]	2200	0.12	<0.09	1360	220	0.00076	<0.0002	0.062	0.091	0.14
AVERAGE ± 1σ	—	0.16±0.16	--	3300±3500 (excluding M-101)	--	0.0054±0.011 (excluding TP-AG)	--	0.098±0.13	--	--

(a) Corrosion film scrapings from Indian Point-1 steam generator
(b) Regenerative heat exchanger piping from Humboldt Bay
(c) Corrosion film deposits on HEPA filter used in steam generator repair
(d) Reactor water cleanup piping from Monticello
(e) Alumina grit used for decontamination of Turkey Point Unit 3 steam generator
(f) Reactor steam vent piping from Dresden-1

TABLE 5.3. Activity Ratios Normalized to $^{239-240}Pu$ in Piping Exposed to Primary Coolant (decay corrected to shutdown date or sampling date)

	$^{137}Cs/^{239-240}Pu$	$^{90}Sr/^{239-240}Pu$	$^{94}Nb/^{239-240}Pu$	$^{55}Fe/^{239-240}Pu$ (x10^5)	$^{63}Ni/^{239-240}Pu$ (x10^3)	$^{99}Tc/^{239-240}Pu$	$^{129}I/^{239-240}Pu$	$^{241}Am/^{239-240}Pu$	$^{244}Cm/^{239-240}Pu$
IP-H27	330	1.5	18	1.5	4.5	0.19	0.057	1.3	0.16
HBRS-26	5	2.2	<2	0.34	0.22	0.11	<0.004	1.9	0.65
RSW-2	237	<36	<4	1.0	62	<0.4	<0.002	0.80	0.69
M-101	1900	36	<9	0.0043	0.035	0.25	<0.008	2.3	7.0
TP-H7	3.1	0.67	<2	0.24	0.080	4.9	<0.01	1.4	0.80
D-Sc-38	17	2.0	<1	0.23	3.4	0.03	<0.003	1.5	2.3
AVERAGE ± 1σ		8.5 ± 15		0.66 ± 0.57		0.98 ± 1.9		1.5 ± 0.5	1.1 ± 0.8

(a) Corrosion film scrapings from Indian Point-1 steam generator
(b) Regenerative heat exchanger piping from Humboldt Bay
(c) Corrosion film deposits on HEPA filter used in steam generator repair
(d) Reactor water cleanup piping from Monticello
(e) Alumina grit used for decontamination of Turkey Point Unit 3 steam generator
(f) Reactor steam vent piping from Dresden-1

6.0 IMPLICATIONS FOR DECOMMISSIONING

This research program has generated a data base describing residual radionuclide concentrations and distributions throughout the operating systems of seven light water reactors. The results of this study have implications for future decommissioning efforts, waste disposal of decommissioning wastes, and additionally point out areas where data base deficiencies still exist. The long-lived radionuclides, including fission products and activated corrosion products, have been examined in operating systems. The data base describes typical concentration ranges observed, and scales the significance of the various radionuclides. The radionuclides examined include a more comprehensive spectrum than included in previous decommissioning assessments (1-5).

Using the data base generated in this program, which describes typical concentration ranges and radionuclide ratios throughout operating systems, more informed decisions can be made regarding four potential decommissioning scenarios: 1) decontamination and decommissioning, 2) immediate decommissioning, 3) safe storage for a period of years followed by decommissioning, or 4) entombment.

6.1. WASTE CLASSIFICATION AND DISPOSAL

The results of this program indicate where typical decommissioning wastes will fall within the present waste classification and disposal guidelines. Section 10, Part 61, Subpart D of the Code of Federal Regulations describes the technical requirements for land disposal facilities and near surface disposal of radioactive waste. Included therein are waste classifications and disposal requirements. The least restrictive waste classification is designated Class A. The limitations for Class A wastes are specified in paragraph 55 of 10 CFR 61 and are listed in curies per cubic meter of waste for specific radionuclides of concern. For more direct comparison with our measured results, these limits are shown in Table 6.1, converted to μCi per cubic centimeter in column 1, and μCi/g of steel in column 2. Columns 3 through 8 list the highest concentrations observed at the reactor sites sampled during this program (excluding Pathfinder which had atypically low residual radionuclide levels). At none of the stations were radionuclide concentrations in corrosion films in contaminated hardware or piping above the limits specified for Class A wastes. Based upon this rather limited sampling, it would appear that all contaminated piping and hardware, excluding the pressure vessel and internals, could be disposed of as the least restrictive class of waste. This would greatly simplify the disposal of all of the contaminated piping and hardware slated for eventual dismantlement during decommissioning. An alternative, and perhaps less restrictive, method is waste classification on a volume basis, rather than the mass basis used here. For pieces of contaminated material having a large volume-to-mass ratio, the licensee would probably prefer to make the classification on a volume basis.

None of the radionuclides having half-lives less than 5 years, either individually or summed together, came close to the Class A limit. The

TABLE 6.1. Comparison of Disposal Limits for Class A Waste with Radionuclide Concentrations Observed in Contaminated Primary Loop Piping

Radionuclide	Class "A" Limit (μCi/cm^3)	Class "A" Limit Converted To μCi/gm of Iron or Steel	Highest Observed Steel Contamination - μCi/gm Humboldt Bay(1)	Dresden One(2)	Monticello(3)	Indian Point Unit One(4)	Turkey Point Unit 3*(5)	Rancho Seco(6)
Any with half-life less than 5 years	700	89	13 (^{55}Fe)	0.86 (^{55}Fe)	0.7 (^{65}Zn)	1.8 (^{65}Zn, ^{55}Fe)	2.9 (^{55}Fe, ^{58}Co ^{65}Zn, ^{54}Mn)	0.03 (^{58}Co, ^{55}Fe)
^{60}Co	700	89	1.9	0.95	0.09	0.1	0.92	0.003
^{59}Ni	22	2.8	$<5 \times 10^{-6}$	0.002	$<6 \times 10^{-6}$	2×10^{-4}	1.5×10^{-4}	6×10^{-5}
^{63}Ni	3.5	0.44	0.08	0.09	4×10^{-4}	0.032	4.5×10^{-3}	0.010
^{90}Sr	0.04	0.0051	9×10^{-4}	5×10^{-5}	2×10^{-4}	8×10^{-6}	3.0×10^{-5}	$<4 \times 10^{-6}$
^{94}Nb	0.02	0.0025	$<9 \times 10^{-4}$	$<3 \times 10^{-5}$	<0.001	$<5 \times 10^{-6}$	$<8 \times 10^{-5}$	$<2 \times 10^{-6}$
^{99}Tc	0.3	0.038	5×10^{-5}	3×10^{-7}	3×10^{-6}	1×10^{-6}	3.0×10^{-4}	$<2 \times 10^{-6}$
^{129}I	0.008	0.001	$<2 \times 10^{-6}$	$<7 \times 10^{-8}$	$<1 \times 10^{-7}$	$<6 \times 10^{-4}$	$<6 \times 10^{-7}$	$<4 \times 10^{-9}$
^{137}Cs	1.0	0.13	0.002	4×10^{-4}	0.02	0.006	$<2 \times 10^{-4}$	$<3 \times 10^{-6}$
Alpha Emitting Transuranics	10 nCi/g	10 nCi/g	1.8	0.16	0.08	0.004	2.2×10^{-4}	0.001
^{242}Cm	2000 nCi/g	2000 nCi/g	9.2	0.003	0.011	N.D.	2.2×10^{-5}	8×10^{-4}

(1) HBRS-26; Regen. Heat Exchanger Piping; decay corrected to shutdown date, July, 1976
(2) D-SC-18; Reactor Steam Vent Piping; decay corrected to shutdown date, October, 1978
(3) M-101; Reactor Water Cleanup Piping; decay corrected to sampling date, November, 1981
(4) IP-H-12; Reactor Water Cleanup Piping; decay corrected to shutdown date, October, 1974
(5) TP-H-7*; Alumina grit from steam generator decontamination; decay corrected to sampling date, October, 1981
(6) RS-H-8A; Letdown cooler piping; decay corrected to sampling date, March, 1982

N.D. - Not Detectible

*Since no primary loop piping was available at Turkey Point, radionuclide concentrations (μCi/gm) associated with an entire Turkey Point Unit 3 steam generator were estimated by dividing the total estimated curies of ^{60}Co (675 curies) in the steam generator by the entire weight of the steam generator (2.0×10^8 gm), and using the ratio of ^{60}Co to the other radionuclides measured in the sample of alumina grit (TP-H-7) from the decontamination of this steam generator.

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regenerative heat exchanger piping at Humboldt Bay came the closest, but was still a factor of about 7 below the limit at the time of reactor shutdown (July, 1976). The long-lived radionuclides, including ^{60}Co, likewise did not come close to the Class A limit. The ^{63}Ni in the most contaminated piping from Humboldt Bay and Dresden-1 came closest to the limit, but were still about a factor of 5 below it. For the alpha emitting transuranic radionuclides, the regenerative heat exchanger piping at Humboldt Bay came closest to the limit, but was still a factor of 5.5 below it.

The steam generators in the PWR's examined in this study contained approximately 77 to 95% of the total residual radionuclides deposited outside of the pressure vessels at these plants. Therefore, it is of interest to determine if the radionuclide concentrations (in μCi/gm) within an entire steam generator would be below the Class A waste limit. If so, the entire steam generators could be disposed of as the least restrictive waste class. Table 6.2 lists the concentrations of 10 CFR 61 radionuclides estimated in entire individual steam generators at Indian Point-1, Turkey Point-3, and Rancho Seco. These concentrations were obtained by dividing the total individual radionuclide inventory within a steam generator by the total weight of the steam generator. The individual steam generator weights were 7.5×10^4 kg, 2.0×10^5 kg, and 5.2×10^5 kg, respectively, for Indian Point-1, Turkey Point-3, and Rancho Seco.

As shown in Table 6.2, all of the radionuclide concentrations in the Indian Point-1 and Turkey Point-3 steam generators were below Class A limits. The only radionuclide whose concentration was above the Class A limit in the Rancho Seco steam generator was ^{63}Ni; however, it was only 1.7 times higher than the Class A limit. Rancho Seco has an unusually high relative amount of ^{63}Ni in the primary loop corrosion deposits due to the extensive use of inconel (70-80% Ni) in primary loop construction materials. It appears that some degree of decontamination of the Rancho Seco steam generators would be necessary to dispose of them as Class A waste.

All of the nuclear stations examined had contaminated concrete that would have to be disposed as low-level radioactive waste if the sites were to be restored to unrestricted use following decommissioning. To evaluate the possible waste disposal options, the following comparisons have been made.

The Class A limits converted to μCi/g for contaminated concrete, based upon an assumed density of 2.5 gm/cm^3 for concrete, are shown in Table 6.3. Also shown in the table are the highest radionuclide concentrations observed in concrete from the reactor stations sampled during this field program. The concentrations listed for each station are based upon concrete cores containing the highest concentration levels for that site. At Indian Point Station-Unit One ^{137}Cs concentrations in several concrete cores from the sump floor of the reactor building were observed to exceed the specified Class A limits by about a factor of 3.5. As pointed out in Section 4.1.4, concrete has a particularly high affinity for ^{137}Cs, since the cesium can irreversibly ion exchange onto the minerals in the

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TABLE 6.2. Comparison of Disposal Limits for Class "A" Waste with Radionuclide Concentrations for Entire Steam Generators

Radionuclide	Class "A" Limit Converted to µCi/gm of Steel	Radionuclide Concentration (µCi/gm) for Entire Steam Generator		
		Indian Point-1(A)	Turkey Point-3(B)	Rancho Seco(C)
Any with half-life less than 5 years	89	2.4 (^{55}Fe, ^{65}Zn ^{54}Mn, ^{134}Cs)	2.9 (^{55}Fe, ^{58}Co ^{65}Zn, ^{54}Mn)	2.3 (^{55}Fe, ^{58}Co ^{54}Mn)
^{60}Co	89	0.43	0.92	0.71
^{59}Ni	2.8	4.9×10^{-4}	1.5×10^{-4}	4.8×10^{-3}
^{63}Ni	0.44	0.057	4.5×10^{-3}	0.76
^{90}Sr	0.0051	2.1×10^{-5}	3.0×10^{-5}	$<5 \times 10^{-4}$
^{94}Nb	0.0025	2.5×10^{-5}	$<8 \times 10^{-5}$	$<1 \times 10^{-4}$
^{99}Tc	0.038	2.4×10^{-6}	3.0×10^{-4}	$<1 \times 10^{-5}$
^{129}I	0.0010	7×10^{-7}	$<6 \times 10^{-7}$	$<7 \times 10^{-7}$
^{137}Cs	0.13	4.6×10^{-3}	$<2 \times 10^{-4}$	0.013
Alpha Emitting TRU	10 nCi/gm	0.042	2.2×10^{-4}	5.4×10^{-5}
^{242}Cm	2000 nCi/gm	N.D.	2.2×10^{-5}	7.7×10^{-5}

(A) At Shutdown October, 1974

(B) As of October, 1981

(C) As of March, 1983

TABLE 6.3. Comparison of Disposal Limits for Class A Waste with Radionuclide Concentrations Observed in Contaminated Concrete

	Class "A" Limit (μCi/cm^3)	Class "A" Limit Converted to μCi/g Concrete	Humboldt Bay[1]	Dresden One[2]	Monticello[2]	Indian Point One[4]	Turkey Point Three[5]	Rancho Seco[6]
Any with half-life less than 5 years	700	280	1.6 (^{55}Fe)	0.13 (^{134}Cs ^{55}Fe, ^{54}Mn)	0.00024 (^{65}Zn ^{106}Ru, ^{59}Mn, ^{134}Cs)	8.3 (^{134}Cs, ^{144}Ce)	0.016 (^{55}Fe, ^{65}Zn, ^{134}Cs)	0.00077 (^{134}Cs)
^{60}Co	700	280	0.016	0.21	0.0034	0.029	0.007	2.9×10^{-4}
^{59}Ni	22	8.8	$\sim3\times10^{-5}$	0.001	N.M.	6×10^{-5}	1×10^{-4}	N.M.
^{63}Ni	3.5	1.4	0.003	0.3	N.M.	0.004	3×10^{-5}	N.M.
^{90}Sr	0.04	0.016	6×10^{-5}	0.001	N.M.	6×10^{-5}	7×10^{7}	N.M.
^{94}Nb	0.02	0.0080	$<2\times10^{-6}$	$<1\times10^{-5}$	$<1\times10^{-6}$	1×10^{-6}	$<2\times10^{-5}$	N.M.
^{99}Tc	0.3	0.12	$<2\times10^{-5}$	1×10^{-6}	N.M.	3.4×10^{-7}	9×10^{-7}	N.M.
^{129}I	0.008	0.0032	$<1\times10^{-7}$	$<1\times10^{-7}$	N.M.	2.7×10^{-7}	$<1\times10^{-7}$	N.M.
^{137}Cs	1.0	0.40	0.005	0.10	2×10^{-4}	1.4	0.01	0.0056
Alpha Emitting Transuranics Half-life >5 yrs	10 nCi/g	10 nCi/g	0.009	0.07	N.M.	5×10^{-4}	3×10^{-4}	N.M.
^{242}Cm	2000 nCi/g	2000 nCi/g	N.D.	0.78	N.M.	N.D.	2.5×10^{-5}	N.M.

(1) HBCC-10, RX Bldg., -66'; HBCC-6, RW Bldg.; decay corrected to shutdown date, July, 1976.
(2) DCC-4, Sub-pile Rm.; decay corrected to shutdown date, October, 1978.
(3) MCC-8, RWCVPR; decay corrected to sampling date, November, 1981.
(4) IP-CC-5, IP-CC-6, IP-CC-9; decay corrected to shutdown date, October, 1974.
(5) TPCC-9, TPCC-12, TPCC-15; decay corrected to sampling date, October, 1981.
(6) RS-CC-1; decay corrected to sampling date, March, 1983.

N.D. = non-detectible
N.M. = not measured

concrete. The contaminated concrete cores collected at the other stations did not exceed the Class A limit. However, since very few contaminated sump areas were sampled, particularly at the operating plants, it is likely that some contaminated concrete areas from other plants will exceed the Class A limit for ^{137}Cs. However, the disposal guidelines are written such that highly contaminated concrete could be disposed as Class A waste if it were diluted with concrete having lower residual radionuclide concentrations, such that the average concentration per disposal container does not exceed the Class A limit.

Two alternative approaches could be utilized to obtain an average concentration below the Class A limit: 1) surficial concrete from areas having radionuclide concentrations above Class A limits could be mixed with concrete (or some other material) with lower residual radionuclide concentrations such that an overall average concentration below the Class A limit is achieved, or 2) since the residual radionuclide contamination of concrete has been noted to decrease typically several orders of magnitude in the first two centimeters, removal of several (4 to 6) centimeters of concrete from the contaminated surface would result in an average radionuclide concentration below the Class A limit, even if the surface (top 2 cm) was well above the Class A limit.

6.2 Allowable Residual Levels of Soil Contamination

The reactor stations examined during this research program typically had some areas within the confines of the site where low concentrations of radionuclides were present in surface soils at levels above those contributed by global fallout. These concentrations typically ranged from less than 1 to over 100 pCi/g of soil or sediment (see Table 4.7). A suggested allowable residual contamination level (ARCL) specifies that the total concentrations of radionuclides do not contribute to a dose rate exceeding 10 mrem/yr to the maximally exposed individual.(19) However, the conversion from concentration in pCi/g to dose in mrem/yr is not a straightforward one. Currently, there are several groups actively developing methodology which allows this conversion. These include groups at the Pacific Northwest Laboratory(19) and at Oak Ridge National Laboratory.(20) Basically, the methodology defines a model which allows calculation of permissible residual contamination levels for any mixture of radionuclides as measured from field samples. The methodology utilizes the physical characteristics of the individual site along with the possible scenarios for human exposure and performs a resultant dose analysis. Conservative assumptions are made which result in the highest potential dose. Thus, a conservative estimate (high) is obtained for any radionuclide mixture. Once the characteristics of the contaminated site and most significant exposure pathway are incorporated mathematically, the allowable residual concentrations can be back-calculated for any radionuclide mixture. Kennedy et al.(19) have performed an analysis to determine the allowable residual contamination in soils at the Shippingport reactor site following decommissioning. Table 6.4 lists the allowable residual contamination level values for individual radionuclides in soil based upon a specific scenario in which contaminated surface soil (0-1 m)

TABLE 6.4. Allowable Residual Contamination Level Values for Individual Radionuclides in Soil at the Shippingport Atomic Power Station Based on 10 mrem/yr to the Maximally-Exposed Individual (from Kennedy, et al.(19))

Radionuclide	Unconfined Soil 0-1 m Deep (pCi/g)(a)
^{3}H	5.9E+7(b)
^{14}C	1.2E+5
^{54}Mn	5.9E+0
^{55}Fe	1.0E+5
^{57}Co	4.4E+4
^{60}Co	9.1E-1
^{59}Ni	2.3E+2
^{63}Ni	1.9E+1
^{90}Sr+D(c)	9.1E-1
^{93}Mo	1.7E+3
^{99}Tc	2.6E+1
^{106}Ru	4.6E+1
^{110m}Ag	1.9E+0
^{124}Sb	3.7E+2
^{125}Sb	5.9E+0
^{134}Cs	1.8E+0
^{135}Cs	1.1E+3
^{137}Cs+D	3.7E+0
^{144}Ce+D	7.7E+2
^{152}Eu	2.0E+0
^{154}Eu	1.8E+0
^{235}U+D	2.0E+1
^{238}U+D	2.3E+1
^{237}Np+D	1.4E+1
^{238}Pu	1.6E+2
^{239}Pu	1.1E+2
^{240}Pu	1.1E+2
^{241}Pu	2.8E+4
^{241}Am	7.1E+1

(a) Based on the residential/home-garden scenario.
(b) Where 5.9E+7 = 5.9 X 10^{7}
(c) +D means plus short-lived daughters

at the decommissioned Shippingport Station was present after the site was converted into a residential development/home-garden situation.

It is of interest to compare the allowable residual contamination levels specified in this scenario with the levels of radionuclide contamination in soils at the seven nuclear plants examined in this study.

For the contaminated soils observed in this study, ^{60}Co was the limiting radionuclide for the allowable residual contamination level. By comparing the observed contamination levels given in Table 4.7 with the calculated allowable residual contamination levels given in Table 6.4, it is obvious that the ^{60}Co concentrations in the most contaminated soils at all of these sites except Monticello exceeded the suggested limit of 0.91 pCi/gm specified in Table 6.4.

Several alternative remedial actions could be performed to reduce the radionuclide concentrations in the contaminated soils to levels which would not contribute to a dose rate greater than 10 mrem/yr. First, the contaminated surface soil could be removed and disposed as low-level waste. Since the extent of soil contamination at the reactor sites was very limited, this may not be an unreasonable alternative. Second, the contaminated surface soils may be plowed under and mixed with uncontaminated soils to give a concentration below the specified limit. This alternative would be feasible if the contamination was confined to the top several centimeters of soil and the contamination was low enough to allow adequate dilution by clean subsurface soil. A third possible alternative would be to provide some sort of institutional control of the site until radionuclide concentrations decayed to levels below specified limits.

7.0 ADDITIONAL RESEARCH NEEDS

The results of this research program indicate several areas where additional information would be desirable. The additional needs can be placed into three general categories: 1) expansion of the present data base, 2) effects and implications of recent innovations in operations parameters and materials, and 3) attention to specific problem areas.

7.1 Expansion of the Data Base

Although this research program examined seven reactor sites, data from additional reactor sites would be extremely desirable for several reasons. First, as noted in the discussion above, the Monticello inventory and radionuclide composition were somewhat anomalous in comparison to the other reactors examined during this research program. It was the only relatively recent BWR examined, thus it is not possible to say whether Monticello is representative of newer BWR's, and that BWR's will generally have significantly lower residual radionuclide inventories. It would be highly desirable to examine several other recent BWR's in order to gain further information regarding BWR inventories.

Second, no gas-cooled reactors were examined in this research program, since this reactor type is rare in the U.S. However, there is increasing interest in this technology at the present time. Thus, it would be worthwhile to examine the residual inventories associated with this reactor type. A low residual inventory might be an additional factor in favor of this reactor type.

Third, at the seven reactor sites examined in this program, there were significant differences observed in radionuclide compositions and residual inventories. Additional site examinations would reinforce or refute these generalizations. Generic statements hypothesized as a result of this research are somewhat tenuous. As an example, it would be useful to determine if the relationship proposed between reactor size and years of service versus total inventory is indeed appropriate over a larger population of reactors.

Fourth, although this study has greatly contributed to a better knowledge of residual radionuclide inventories within nuclear power plants, it was also subject to limitations imposed by the operational or standby status of the reactors which were examined. It was concluded that optimum information could be best obtained during an actual reactor decommissioning, during which adequate samples for radionuclide characterization would be available. For example, the upcoming decommissioning and dismantling of the Shippingport Station offers the best opportunity to date to carry out such a study. Being able to take advantage of this unique opportunity to follow up the study just completed would provide a detailed radiological assessment, and would help resolve some of the unanswered questions concerning the radiological problems encountered during an actual decommissiong.

7.2 Effects of Recent Trends and Innovations

The possible trend as noted at Monticello toward lower residual inventories is only one of several potential changes in residual radionuclide inventories. Radionuclide compositions are also subject to change with the introduction of newer materials. Two examples of this effect are: 1) the relatively large amounts of ^{63}Ni in the Rancho Seco inventory produced by more extensive use of inconel in the reactor systems, and 2) the large amount of ^{65}Zn at Monticello due to the presence of admiralty brass heat exchangers. Utilization of admiralty brass in reactor systems is decreasing because of the tendency toward corrosion, with the consequent production of large amounts of ^{65}Zn. In contrast, the increased relative amounts of ^{63}Ni and perhaps ^{110m}Ag at Rancho Seco may reflect the use of newer materials, with both lower stable cobalt concentrations and lower corrosion rates. These newer materials are being employed to decrease production of ^{60}Co, the largest contributor to dose rates in operating systems, as well as corrosion product formation and translocation throughout reactor systems.

There are several other trends and innovations in reactor operations which will affect the residual radionuclide distribution and inventories at operating nuclear stations. These include improved controls on water chemistry and high temperature or electromagnetic filtration of primary loop water. Improved control of water chemistry tends to minimize corrosion processes and subsequent neutron activation and redistribution of the activated corrosion product materials throughout operating systems. High temperature and/or electromagnetic filtration for reactor water cleanup in primary loops of PWRs, when fully commercialized, may decrease radionuclide inventories in primary loops of PWRs, and especially in the steam generators where the largest fraction of the primary loop residual inventory resides. A final factor which may have the most significance upon residual contamination inventories may be the growing practice of periodic primary system chemical decontamination.

References

1. Smith, R. I., G. J. Konzek and W. E. Kennedy, Jr., "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," NUREG/CR-0130, Vols. 1 and 2. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, June, 1978.

2. Oak, H. D., G. M. Holter, W. E. Kennedy, Jr., and G. J. Konzek, "Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station," NUREG/CR-0672, Vols. 1 and 2. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, June, 1980.

3. Manion, W. J. and T. S. LaGuardia, "Decommissioning Handbook," DOE/EV/-10128-1. Prepared for the U.S. Department of Energy by Nuclear Energy Services, Incorporated, Automation Industries, Danbury, Connecticut, November, 1980.

4. U.S. Nuclear Regulatory Commission, "Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," NUREG-0586, U.S. Nuclear Regulatory Commission, Office of Standards Development, Washington, D.C., 1981.

5. Osterhout, M. M., ed. Decontamination and Decommissioning of Nuclear Facilities, Plenum Press, New York, 1980.

6. Robertson, D. E., C. W. Thomas, K. H. Abel, J. C. Evans, M. W. Leale, E. A. Lepel, and E. M. Woodruff, "Topical Report-Residual Radionuclide Distribution and Inventory at the Pathfinder Generating Plant," PNL-4326. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, June, 1982.

7. Abel, K. H., D. E. Robertson, C. W. Thomas, E. A. Lepel and V. W. Thomas, "Residual Radionuclide Distribution and Inventory at the Humboldt Bay Nuclear Plant, PNL-4628. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, May, 1983.

8. Robertson, D. E., C. W. Thomas, K. H. Abel, E. A. Lepel, W. V. Thomas, L. C. Carrick, and J. C. Evans, "Residual Radionuclide Distribution and Inventory at the Monticello Nuclear Generating Plant," PNL-4794. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, June, 1983.

9. Lepel, E. A., D. E. Robertson, K. H. Abel, W. V. Thomas, J. C. Evans, and M. W. Leale, "Residual Radionuclide Distribution and Inventory at Indian Point One," PNL-5020. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, April, 1984.

10. Abel, K. H., D. E. Robertson, E. A. Lepel, and M. W. Leale, "Residual Radionuclide Distribution and Inventory at the Dresden Unit One Nuclear Generating Plant," PNL-4961. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, December, 1983.

11. Abel, K. H., D. E. Robertson, J. C. Evans, and L. C. Carrick, "Residual Radionuclide Distribution and Inventory at Turkey Point Power Station Units No. 3 and 4," PNL-5010. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, December, 1984.

12. Abel, K. H., D. E. Robertson, E. A. Lepel and L. C. Carrick, "Residual Radionuclide Distribution and Inventory at Rancho Seco Nuclear Generating Station," PNL-5146. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, June, 1984.

13. Evans, J. C., E. A. Lepel, R. W. Sanders, C. L. Wilkerson, W. B. Silker, C. W. Thomas, K. H. Abel, and D. E. Robertson, "Long-Lived Activation Products in Reactor Materials," PNL-4824, NUREG/CR-3474. Prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, 1984.

14. U.S. Nuclear Regulatory Commission, "Licensing Requirements for Land Disposal of Radioactive Wastes," 10 CFR 61. Federal Register, Vol. 46, No. 142, Friday, July 24, 1981.

15. Kennedy, W. E. Jr., B. A. Napier and J. K. Soldat, "Allowable Residual Contamination Levels in Soil for Decommissioning the Shippingport Atomic Power Station Site," PNL-4801. Prepared for the U.S. Department of Energy by Pacific Northwest Laboratory, Richland, WA., September, 1983.

16. Simmons, G. L., et al., "ISOSHLD II Code Revision to Include Calculation of Dose Rates from Shielded Bremsstrahlung Sources," BNWL-236, Supplement 1, Pacific Northwest Laboratory, Richland, Washington, 1967.

17. Bergmann, C. A., J. Roesmer and D. W. Perone, "Primary-Side Deposits on PWR Steam Generator Tubes," NP-2968. Prepared by Westinghouse Electric Corporation for the Electric Power Research Institute, Palo Alto, CA, March, 1983.

18. Helmholz, H. R., et al., "Solid Radwaste Radionuclide Measurements," NP-2734, Electric Power Research Institute, Palo Alto, California, November, 1982.

19. Kennedy, W. E., Jr., B. A. Napier and J. K. Soldat, "Allowable Residual Contamination Levels in Soil for Decommissioning the Shippingport Atomic Power Station Site," PNL-4801, Pacific Northwest Laboratory,

Richland, Washington, September, 1983; Napier, B. A., "A Method for Determining Allowable Residual Contamination Levels of Radionuclide Mixture in Soil," PNL-3852, Pacific Northwest Laboratory, Richland, Washington, 1982.

20. Eckerman, K. F. and M. W. Young, "A Methodology for Calculating Residual Radioactivity Levels Following Decommissioning," NUREG-0707, U.S. Nuclear Regulatory Commission, Washington, D.C., 1980.

APPENDIX A

APPENDIX A

ON-SITE AND LABORATORY SAMPLING AND ANALYSIS METHODS

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APPENDIX A
ONSITE AND LABORATORY SAMPLING AND ANALYSIS METHODS

A.1. SAMPLING PROGRAMS AT THE NUCLEAR STATIONS

Sampling programs for each of the sites were individually designed, based upon the configuration of the station, its operational status, and sample availability. Sampling at each site focused upon acquisition of contaminated equipment, hardware, or corrosion films from operational systems, as well as concrete from contaminated areas within the station, and soils and sediments from within the exclusion area of the station. The sample inventories for the field program are summarized in Table A.1. The table details the number and types of samples, e.g., hardware, soils, and concrete obtained during the onsite sampling and measurements programs.

The miscellaneous column contains those samples which do not fit elsewhere in the table. Examples are a HEPA filter from Rancho Seco which was associated with repairs on the upper primary side of one of the steam generators, as well as samples of alumina grit from Turkey Point which were generated during the decontamination of the steam generators prior to removal. A complete listing and description of samples obtained during the field sampling operations phase of this research program are contained in Appendix B. Appendix B also contains a listing by site and sample type, as well as other documentation regarding the samples, such as dose rate and origin.

TABLE A.1. Sample Inventory from the Nuclear Power Stations Examined

Station	Piping and Hardware	Corrosion Film Scrapings	Concrete Cores	Misc.	Soil
Pathfinder	52	0	22	6	10
Humboldt Bay	5	12	27	12	16
Indian Point-1	8	10	15	3	0
Dresden-1	2	12	8	2	8
Monticello	9	2	10	1	4
Turkey Point	4	6	16	2	11
Rancho Seco	4	5	2	1	8

A.2. SAMPLING PROCEDURES

A sampling procedure was developed for each type of sample obtained at the reactor sites. Sample types included hardware, piping, corrosion films, concrete, soils and sediments. The individual procedures for each type of sample are described below.

A.2.1. Hardware Samples

Components from operating systems made available for our sampling purposes were sampled in one of four ways. The first option, if the component was small enough, was taking the whole item intact. Alternatively, if this was impractical, the component was subsampled in one of three ways. A corrosion film specimen was obtained in those instances where continued integrity of the component was mandatory. The deposited corrosion film was removed from the surface of interest using a paint or razor blade scraper. A predetermined surface area was scraped and the loosened corrosion film gathered into a polyethylene bag. The bag containing the corrosion film was then sealed, placed into another uncontaminated plastic bag, and the second bag also sealed. The sample was then labeled for future identification with a log number, date of origin, location of the component and its dose rate. It is estimated that this scraping technique removed between 70 to 90% of the deposited corrosion product material.

Two other procedures were utilized where component integrity was not a concern. Most frequently, a subsample was obtained using a portable bandsaw. In this way a segment of pipe or other component could be obtained which was large enough such that the heat generated during the cutting process did not alter the sample. The bandsaw was utilized for cutting pipe up to 4 inches in diameter and for metal plate materials. For plate, a corner was cut, or alternatively, a notch was cut along appropriate edges. Where piping was too large, or the sample inaccessible to the bandsaw, a hole saw was employed. This was a less desirable method since the sample was smaller in size and potentially disturbed by the cutting operation.

A.2.2. Concrete Samples

Concrete was sampled either by coring or chipping. Concrete cores, 10 cm diameter by 6 to 13 cm deep, were collected from each nuclear station. Cores were normally taken from contaminated floors from the main buildings at each plant, including the reactor building, turbine building, auxiliary building, or other contaminated areas. Prior to taking a core, the floor was surveyed with a G-M survey meter to locate "hot spots" of contamination. Reinforcing steel (rebar) was also located using a portable rebar locator prior to coring. In this way cores could be taken from areas free from steel reinforcing bar, since at times concern was expressed by plant personnel that the coring might affect the integrity and strength of the floor.

The coring system utilized a diamond-tipped core barrel and recirculated water as both a bit coolant and lubricant. After coring and cleanup of residual water, the core segment was broken free from the floor, dried, wrapped in rags, enclosed in double plastic, and then labeled. The cores were finally placed into metal cans prior to placement into the shipping drum for return to our laboratory. After coring operations were completed, the remaining hole was dried of excess moisture, refilled with concrete, and the surface finished by trowelling.

At the Pathfinder, Dresden Unit One, Humboldt Bay, and Indian Point Unit One nuclear units, concrete samples were also obtained from the biological shield surrounding the reactor vessel. This sampling was accomplished using a hammer and chisel at the Pathfinder station, or using a portable, powered rotary hammer and spalling bit at Humboldt Bay, Dresden Unit One, and Indian Point Unit One.

A.2.3. Soil and Sediment Samples

Soil samples were usually taken at approximately the four compass directions around the plants and/or at locations where contamination was known or suspected within the exclusion area of the plant. At several plants, soils from more distant locations were also obtained. A large surface area coring device or a shovel was used to remove the soil from a 0.1 square meter area, and the soil was screened through a nominal No. 20 mesh sieve to remove large aggregate. All soil samples were packaged in plastic bags, labeled as to origin and date, and placed into the shipping drum for return to our laboratory.

At most sites, samples of sediment from cooling canals or processed water effluent streams were also obtained. These were collected using a hand-held coring device which sampled the upper 10-15 cm of sediment. Upon sample retrieval, the surficial water was carefully decanted from these samples and the sediment packaged into plastic bags. The sediment was then double bagged, the outer bag appropriately labeled, and the sample placed into the shipping drum for return to the laboratory. For samples of sufficient length (>10 cm), the sediment cores were divided into surficial and lower sections.

A.3. ONSITE MEASUREMENTS

A number of onsite measurements were made at five of the plants using a portable intrinsic germanium gamma-ray spectrometer and a beta-gamma detector. The instrumentation was set up in a low background area at each site, and a lead shield was constructed around the detectors. Samples were then brought into this low background area for preliminary analyses of gamma and beta emitting radionuclides. These measurements served as a guide during the sampling program.

The intrinsic germanium detector (a 30% efficient Princeton Gamma-Tech coaxial diode with a resolution of 1.94 keV FWHM) was coupled to a Canberra Series 80 multichannel analyzer and minicomputer. A magnetic tape deck and hard copy printer were also interfaced to the multichannel analyzer to provide data storage and/or onsite printout of the spectral data.

The beta-gamma detector (Eberline) was used primarily for monitoring smearable contamination levels on floors and equipment. At the Pathfinder site, measurements were made using the intrinsic germanium detector in a portable mode in order to characterize an area of concrete flooring in the fuel cask loading facility which was known to contain residual contamination.

The floor area was subdivided into equal area components and the subareas surveyed using the spectrometer. The detector face was placed vertically over the floor area being surveyed. A cylindrical lead collimator and shield were fastened in place around the detector during the floor survey.

Measurements were also made at the Pathfinder site of hot spots in the secondary piping and condenser area using the portable gamma-ray spectrometer system. The detector, with the cylindrical lead collimator shield in place, was positioned in front of the hot spot and a measurement taken.

A.4. LABORATORY PROCEDURES

Upon return of the samples from the reactor site to the laboratory, the samples were prepared for analyses. These preparations depended upon physical parameters and analytical needs. The samples, where physically possible, were subjected to nondestructive gamma-ray spectrometric analysis prior to any sample dissolution and radiochemical separations. After sample dissolution by acid digestion or further subsampling, aliquots were again analyzed by gamma spectrometry. The procedures for sample preparation are detailed below by sample type.

A.4.1. Hardware

Samples of piping and other reactor components, e.g., baffle plates, valves, tubing, and filters were leached with hot concentrated hydrochloric acid to dissolve corrosion films. A protective plastic coating was applied prior to the hot acid leach, in order to minimize dissolution from metal surfaces which were not contaminated (e.g. exterior walls of piping) since increased concentrations of dissolved stable elements could potentially interfere with the subsequent radiochemical separations.

The acid leaching procedure entailed submersion of the component in fresh aliquots of hot concentrated hydrochloric acid for two 1-hour periods. The acid leached sample was finally subjected to a rinse in hot deionized water for a period of one hour. All leaching and rinse solutions were filtered, combined, and brought to a final volume of 1 liter by evaporation. The final filters were analyzed by gamma spectrometry. Less than 10% of the total sample activity was retained upon the filter. Dose rates and/or count rates on the hardware components were also measured before and after the leaching procedure. Typically, greater than 80-90% of the measured activity was solubilized during the leaching process.

A.4.2. Corrosion Films

Corrosion films scraped from reactor system surfaces were subjected to hot concentrated hydrochloric acid digestion in the same manner as hardware components. For corrosion films this was usually accomplished in a glass beaker on a stirring hot plate. Two dissolution periods of one hour each followed by a rinse in hot deionized water for an additional hour were again used. The solutions were filtered and the filter from the initial leaching steps placed in the succeeding leaching solution. As noted previously, the overwhelming fraction of the activity was removed into the leachate.

A.4.3. Concrete Cores

Concrete cores were repackaged into polyethylene bags after receipt and subjected to gamma spectrometric analysis. The whole cores were analyzed with the original surface of the core facing the detector. The concrete cores were later cut into 1 cm segments with a diamond bladed lapidary saw. After segmenting, each slab was packaged and analyzed for gamma-ray emitting radionuclides in order to determine radionuclide contamination as a function of depth in the concrete.

Selected concrete core segments were subjected to the acid dissolution procedure since analyses were desired for beta, alpha and low-energy photon emitting radionuclides. The dissolution procedure was similar to that utilized for hardware and corrosion film samples. Initial acid addition was performed in small aliquots and at room temperature due to the vigorous reaction with the carbonate fraction of the concrete. Leaching was also conducted in a large (at least 2 liter) beaker to minimize overflow resultant from the foaming which often occurred during dissolution. After the initial vigorous reaction ceased, the samples were heated, the volume brought to 500 ml of acid and the sample refluxed for one hour. As with hardware and corrosion films, successive leaches were conducted, twice with hydrochloric acid followed by a hot deionized water rinse. The solutions were filtered, and the filter leached in the succeeding step. Finally, all solutions were combined and brought to a final volume of one liter by evaporation.

A.4.4. Soils and Sediments

Upon receipt in the laboratory, soil and sediment samples were placed into large evaporating dishes and oven dried at 105° C. After drying, the soils were sieved to pass a 200 mesh screen. An aliquot of the <200 mesh material (typically 200 g) was placed into a standardized geometry and analyzed by gamma-ray spectrometry. Sediments were dried and ground using a mortar and pestle, and sieved. An aliquot of the <200 mesh material was then analyzed in the same geometry used for soils.

A.5. ANALYTICAL MEASUREMENTS PROGRAM

The analytical measurements for all samples were conducted using direct gamma-ray spectrometry and a combination of radiochemical analyses for non-gamma-ray emitting radionuclides of interest. The direct and radiochemical analytical procedures are described below.

A.5.1. Direct Gamma-Ray Spectrometric Measurements

All samples were analyzed using Ge(Li) or intrinsic germanium (IG) detectors and the resultant gamma-ray spectra subjected to a computer-assisted data reduction program. The sample counting intervals were dependent upon overall sample count rates and radionuclide composition. Often, two counting intervals were employed in order to best evaluate both major and trace radionuclide components present in the samples. The radionuclides determined during the analyses are shown in Table A.2, along with analytical parameters such as half-life, gamma-ray employed for analyses, and any potential interferences.

The computer-assisted data reduction program is performed on a comparative basis, using NBS traceable calibrated radionuclide mixtures prepared into the same counting geometries as utilized for the samples. A calibration curve for the specific geometry of interest is prepared using the standard and the absolute disintegration/count (d/c) for the radionuclides in the standard. The counting efficiency for the radionuclides of interest is then determined from the calibration curve for their specific gamma-ray energies. In all cases, Compton corrections are applied, and activities are decay-corrected back to the date of sampling. Volumetric corrections are also applied so that data can be reported in pCi/cm^2 or pCi/g of material, as applicable.

A.5.2. Radiochemical Separations

For those radionuclides of interest not emitting gamma-rays, radiochemical separations followed by measurement of either beta, alpha, or x-rays was necessary. Chemical separations were conducted for the following radionuclides: ^{59}Ni, ^{63}Ni, ^{55}Fe, ^{90}Sr, ^{99}Tc, ^{129}I, ^{238}Pu, $^{239-240}$Pu, ^{241}Am, ^{242}Cm, and ^{244}Cm. Selected samples from Humboldt Bay were analyzed for uranium isotopes, selected samples from Pathfinder for ^{14}C, and selected samples from Indian Point, Turkey Point, and Dresden for ^{237}Np. In all cases, a radionuclide (isotope of the same element) or chemical yield tracer was utilized to obtain accurate quantitative analyses. The radionuclides subjected to radiochemical separations, the yield tracers utilized, the detection method employed, and the specific emission and its associated energy are shown in Table A.3. The detailed chemical separation procedures are contained in this Appendix. After radiochemical purification, the separated radionuclides were analyzed by appropriate instrumentation, e.g. gas proportional detectors, intrinsic germanium detectors optimized for low energy photons, or surface barrier alpha energy detectors. The instrumental methodology is also detailed in this Appendix.

A.5.3 Separation Procedures

The analytical procedures employed for radiochemical separation and purification prior to analytical determination for the radionuclides of interest to this research program are briefly described below. We have also indicated the instrumentation used for final radionuclide

TABLE A.2. Radionuclides Determined by Gamma-ray Spectrometry Nuclear Data Relating to Spectrometric Analysis

Radionuclide	$t_{1/2}$ (yrs)	Gamma-ray (Kev)	Gamma-ray/Disint.	Alternate Gamma-ray	Alt. Gamma/Disint.	Potential Interferents
Na-22	2.60	1275	0.90			Eu-154
Cr-51	0.076	320	0.095			
Mn-54	0.855	835	1.0			
Co-57	0.740	122	0.856			Eu-152, Eu-154
Co-58	0.194	811	0.994			
Co-60	5.27	1173	0.999	1332	1.0	
Zn-65	0.668	1115	0.498			Eu-152, Sc-46
Nb-94	20,000	703	1.0			Ag-110m
Ru-103	0.108	497	0.9	610	0.055	
Ru-106	1.0	622	0.099			
Ag-108m	127	434	0.90			
Ag-110m	0.686	937	0.343	658	0.944	
Sb-124	0.165	1691	0.50			
Sb-125	2.77	428	0.296	177	0.063	
Sn-126	100,000	415	0.81			
Cs-134	2.06	796	0.89	605	0.98	
Cs-137	30.1	662	0.846			Ag-110m
Ce-141	0.089	145	0.493			Fe-59
Ce-144	0.780	134	0.108			Se-75
Eu-152	12.4	344	0.314	1408,245	0.243, 0.072	
Eu-154	8.5	996	0.107	1005	0.176	
Eu-155	4.96	105	0.244			Gd-153, Lu-177m
Ho-166m	1200	712	0.550			Sb-124, Eu-154
Ra-228(Ac-228)	5.75	969	0.133			Sb-124
Am-241	433	60	0.353			Eu-155, Dy-159

TABLE A.3. Radionuclides Determined After Radiochemical Separations. Description of Yield Tracers, Analytical Instrumentation, and Type of Emission Detected along with Its Associated Energy.

Radionuclide	Yield/Tracer	Detection Method	Type and Energy of Emissior	
C-14	--	Gas Proportional Counter	Beta	0.2 Mev
Fe-55	Stable Fe	Intrinsic Ge Detector	Mn x-ray	5.9 Kev
Ni-59	Stable Ni or Ni-65	Intrinsic Ge Detector	Co x-ray	6.9 Kev
Ni-63	Stable Ni or Ni-65	AC. Shielded Gas Proportional Counter	Beta	0.07 Mev
Sr-90	Sr-85	Gas Proportional Counter	Beta	0.5 Mev
Tc-99	Tc-95m	Gas Proportional Counter	Beta	0.3 Mev
I-129	I-131	Intrinsic Ge Detector	Xe x-ray	0.3 Mev
Pu-238	Pu-242	Alpha Energy Spectrometer	Alpha	5.5 Mev
Pu-239	Pu-242	Alpha Energy Spectrometer	Alpha	5.2 Mev
Am-241	Am-243	Alpha Energy Spectrometer	Alpha	5.5 Mev
Cm-242	Am-243	Alpha Energy Spectrometer	Alpha	6.1 Mev
Cm-244	Am-243	Alpha Energy Spectrometer	Alpha	5.8 Mev

quantification as well as the internal tracer utilized for quality assurance and yield determination.

Carbon-14

Carbon-14 is separated by either acid distillation in the case of inorganic carbonates or by oxidation at high temperature in the case of organic carbon compounds or carbides. The distilled carbon dioxide is trapped in either instance in a caustic solution. Analytical yields are determined gravimetrically using a barium carbonate precipitate. Carbon-14 is quantified using a windowless beta proportional counter, and beta absorption curves are determined on all samples to confirm the carbon-14 measurement.

Iron-55

The analytical procedure utilized for iron-55 entails initial separation by precipitation as the hydroxide in the presence of stable iron carrier and an iron-59 yield tracer. The hydroxide is then dissolved into strong hydrochloric acid and the solution passed through an anion exchange column, where the iron chloride complex is retained. Iron is eluted from the exchange media using strong nitric acid. This solution is evaporated to dryness, the residue dissolved in acid, and the iron electroplated from an oxalate-sulfate media onto a copper disc. The iron-55 is quantified using a thin window intrinsic germanium diode via the Mn x-ray. Analytical yields are determined simultaneously using an iron-59 gamma-ray.

Nickel-59 and Nickel-63

The nickel separation entails initial precipitation of the hydroxide and additional purification using dimethylglyoxime. After destruction of the nickel dimethylglyoxime separate, the nickel is electroplated onto a stainless steel disc from a basic sulfate solution. Nickel-59 is quantified using a thin window intrinsic germanium diode via the cobalt x-ray emitted during decay. Nickel-63 is determined using a NaI(Tl) anticoincidence shielded windowless beta proportional counter. Absorption curves are determined for all samples to confirm the nickel-63. Nickel-65 is utilized as an internal tracer for quality assurance and yield determination.

Strontium-89 and Strontium-90

Strontium isotopes are separated by consecutive precipitation of the basic carbonate followed by precipitation in fuming nitric acid. After an ingrowth period, the ^{90}Sr is then calculated from the ^{90}Y daughter measurements. Strontium-89 is determined by difference. A measurement of total strontium (^{90}Sr and ^{89}Sr) is made immediately after separation via fuming nitric acid. The ^{90}Sr determined after ^{90}Y ingrowth is subtracted from the total strontium measurement to give ^{89}Sr. The yield for the ^{90}Y separation after ingrowth is determined gravimetrically.

Niobium-94

Niobium was separated from other radionuclids by precipitation of niobic oxide from an acid medium during dissolution of the solid sample. Both niobium carrier and ^{95}Nb were present during dissolution. Niobium-94 was measured by gamma-ray spectrometric techniques and radiochemical yield determined by tracing with ^{95}Nb.

Technetium-99

The technetium-99 separation procedure entails initial purification using repeated co-precipitation with iron hydroxide followed by separation onto anion exchange resin. The technetium is eluted in the presence of 9 M ammonium nitrate in a strongly acidic solution. The separated technetium is then electroplated from a basic solution onto a copper disc. Technetium-99 is quantified using a thin window beta proportional detector. Absorption curves are determined for all samples to confirm the presence of technetium-99. Technetium-95 is used for quality assurance and determination of analytical yield during the separation and purification procedure.

Iodine-129

Iodine-129 is initially separated onto anion exchange resin. Elution is achieved by oxidation. The element is then further purified by solvent extraction using carbon tetrachloride and hydroxylamine-hydrochloride. The iodine is then back-extracted into a water/sulfite solution. The iodide is then co-precipitated with cuprous chloride. Iodine-129 is quantified using a thin window intrinsic germanium detector through measurement of the xenon x-ray. Iodine-131 is utilized for quality assurance as a yield tracer during the purification procedure.

Neptunium-237

Neptunium-237 is separated from the sample onto anion exchange resin from a strong nitric acid solution. The anion exchange column is then washed with strong hydrochloric acid containing ammonium iodide. The neptunium is then eluted using hydrochloric-hydrofluoric acid, evaporated to dryness with nitric acid and subsequently electroplated from a weak sulfuric acid solution. The separated and purified ^{237}Np is determined using an alpha energy spectrometer. Neptunium-239 is utilized during the separation procedure for yield tracer and quality assurance purposes.

Plutonium-238 and Plutonium-239/240

Plutonium isotopes are separated by anicn exchange from a strong nitric acid solution. The resin is eluted using hydrochloric acid-ammonium iodide solution. The plutonium is then electroplated from a weak sulfuric acid solution onto a stainless steel disc. Plutonium-242 is used for quality assurance purposes as a yield tracer and plutonium isotopes determined via alpha energy spectrometry. Plutonium-242 is used as a yield tracer for two reasons. First, the alpha emission energy for ^{242}Pu is lower than for ^{238}Pu and $^{239-240}$Pu; thus there is no potential for interference in the analytical peaks of interest. Secondly, the half-life is longer than the other possible tracer ^{236}Pu. Use of ^{242}Pu thus reduces recalibration requirements and decay corrections. This also makes the quality assurance process easier to maintain.

Americium-241, Curium-242, and Curium-244

Because of their very similar chemistries, americium and curium isotopes are separated and purified in one procedure. The isotopes are co-precipitated with iron hydroxide and then dissolved in strong nitric acid. Plutonium and neptunium are removed from the analytical solution by filtering through anion resin. The americium and curium in the filtrate are then co-precipitated at pH 3.0 using 1 mg of calcium carrier and oxalic acid. The americium and curium isotopes are then electroplated from the dissolved oxalate precipitate in a weak sulfuric acid solution onto a stainless steel disc. Americium-243 is used for analytical yield determination and isotopic concentrations determined using alpha energy spectrometry.

APPENDIX B

SAMPLE INVENTORY OF CONTAMINATED PIPING, HARDWARE, CONCRETE, SOILS AND OTHER MISCELLANEOUS SAMPLES COLLECTED AT THE NUCLEAR POWER STATIONS

APPENDIX B

SAMPLE INVENTORY OF CONTAMINATED PIPING, HARDWARE, CONCRETE, SOILS AND OTHER MISCELLANEOUS SAMPLES COLLECTED AT THE NUCLEAR POWER STATIONS

B.1 PATHFINDER B.1
B.2 HUMBOLDT BAY B.6
B.3 DRESDEN UNIT ONE B.14
B.4 INDIAN POINT UNIT ONE B.18
B.5 MONTICELLO B.26
B.6 TURKEY POINT B.30
B.7 RANCHO SECO B.34

B.1 PATHFINDER SAMPLE INVENTORY

PATHFINDER SAMPLE INVENTORY AND DISPOSITION--SAMPLED JULY 1980

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
1	WP-76 300-A	3 1/2" dia. SS pipe from reactor water purification line--end with 1" pipe welded in at 90°--pipe stored in reactor building	8,500 thru pipe 40,000 at open end of pipe	Cut into following pieces 300A-2 300A-1 300A-4 300A-3
2	WP-76 300-B	Same as No. 1--next 6" long piece of straight pipe	40,000 at open end of pipe	Given to J. R. Divine
3	WP-76 300-C	Same as No. 1--next 6" long piece of straight pipe	45,000 at open end of pipe	Archive sample
4	WP-42 No. 11	1 3/4" dia. hole saw plug from 8" dia. pipe from carbon steel reactor feedwater pump suction--pipe stored in reactor building	<200 outside surface	Directly counted on Ge(Li)--leached for radiochemistry
5	WP-74 301-A	2" dia. hole saw plug from 8" dia. pipe from carbon steel reactor feedwater line--pipe stored in reactor building	30,000 at inner surface of pipe 25 through pipe	Directly counted on Ge(Li)--leached for radiochemistry
6	WP-74 301-B	Same as No. 6	50,000 at inner surface of pipe	Directly counted on Ge(Li)--sent to J. R. Divine
7	Concrete Core No. 1	see accompanying listing		
8	Concrete Core No. 2	see accompanying listing		
9	MSB-A	2" dia. hole saw plug from 6" dia. carbon steel pipe from main steam bypass line sampled at second level of steam chase 5' above grating--reactor building		Directly counted on Ge(Li)--leached for radiochemistry
10	MSB-B	Same as No. 9	17,000 at inner surface of pipe	Directly counted on Ge(Li)
11	MSB-C	Same as No. 9	15,000 at inner surface of pipe	Directly counted on Ge(Li)--sent to J. R. Divine

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
12	RFW-A	2" dia. hole saw plug from 8" dia. reactor feedwater line--sampled from 3rd level of steam chase 3' above grating in reactor building	45,000 on inside surface of plug	Directly counted on Ge(Li)--leached for radiochemistry
13	RFW-B	Same as No. 12	55,000 on inside surface of plug	Counted directly on Ge(Li)--sent to J. R. Divine
14	SPC-A	2" dia. hole saw plug from 6" dia. shield pool cleanup line--sampled at Y near hot spot on 3rd level of steam chase in reactor building--heavy corrosion film red on top and yellow on bottom	55,000 on inside surface of plug	Directly counted on Ge(Li)--leached for radiochemistry
15	SPC-B	Same as No. 14	25,000 on inside surface of plug	Directly counted on Ge(Li)--sent to J. R. Divine
16	RLL-A	5" long section of 2 3/8" dia. pipe from reactor liquid level column (lower leg)--from reactor building	100,000; 61 m rad; 1 mR at end of pipe	Not cut up--leached for radiochemistry
17	RLL-B	Same as No. 16	Same as No. 16	Sent to J. R. Divine
18	TR-A	Strip of SS cut from tool rack on south side of bottom of shield pool--reactor building	100 c/m thru plastic bag	Sent to J. R. Divine
19	TR-B	Same as No. 18	1,900 c/m thru plastic bag	One piece 12" long x 3" wide x 1/4" thick cut into four pieces 3" long--also two 3/4" SS nuts and washers labeled TRB-NW
20	SPD-A	End of ~4" dia. SS drain pipe from bottom of shield pool (NE side of shield pool)--reactor building	1,400 c/m thru plastic bag	
21	SPD-B	Same as No. 20	1,500 c/m thru plastic bag	Sent to J. R. Divine
22	FHR-A	2 1/2" dia. SS pipe section from fuel shoot support strut at bottom of shield pool--reactor building	300 c/m thru plastic bag	14" long piece cut into 3 pieces 4" long piece sent to J. R. Divine 4" long piece for leaching for radiochemistry 4" long piece for archives
23	SPC-A	3 1/2" dia. SS pipe from fuel storage pool cleanup line--sampled at basement of fuel handling building--line from bottom of fuel storage basin to filter--demin. cleanup in FHB basement	80,000	Cut into two pieces 2-3" long for leaching for radiochemistry

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
24	SPC-B	Same as No. 23	80,000	Sent to J. R. Divine
25	PWC-A	2" dia. carbon steel from pool water cleanup pump--discharge to Series 11--pool water for storage basin and shield pool to inlet of No. 11 prefilter--sampled from fuel handling building basement	20,000	Sent to J. R. Divine
26	PWC-B	Same as No. 25	20,000	4" long piece--not cut up--leach as is for radiochemistry
27	PDO-A (no B taken)	2 3/8" dia. SS pipe from inlet to demin. for water from storage and shield pools--sampled from fuel handling building basement	2,000	Cut into three pieces 5" long piece for archive 3 1/2" long piece sent to J. R. Divine 3 1/2" long piece for leaching
28	PDI-A (no B taken)	2" dia. SS pipe from inlet to demin. for storage and shield pools--sampled in SE corner of basement of fuel handling building	15,000	Cut into three pieces 4" long piece for archive 3" long piece sent to J. R. Divine 4" long piece cut into two 2" long pieces for leaching
29	SPCB-A (no B taken)	3 1/2" dia. carbon steel pipe from shield pool coolant bypass-water from shield pool to filter demin.--sampled in SE corner of basement of fuel handling building	2,500	Cut into two pieces 3" long piece sent to J. R. Divire 4" long piece for leaching
30	CWTD-A	2" dia. SS pipe from concentrated waste tank discharge line--sampled from basement of fuel handling building	--	4" long piece--not cut up--leach as is
31	CWTD-B	Same as No. 30	--	Sent to J. R. Divine
32	SRTD	1 3/8" dia. x 7" long SS pipe from spent resin tank discharge line--sampled from basement of fuel handling building	--	Cut in half 3 1/2" long piece sent to J. R. Divine 3 1/2" long piece for leaching
33	HSHT	2" dia. SS elbow pipe from high solids holdup tank--just upstream of suction pump--includes resin and black crud trapped in bend--sampled from basement of fuel handling building	--	Cut into two pieces 3" long straight piece sent to J. R. Divine 4" long elbow for leaching--crud removed from elbow and bagged
34	IFDP	3 1/2" dia. x 12" long SS pipe from inlet line to filter--demin. purification system--sampled from basement of fuel handling building	--	Cut into three pieces 3" long piece sent to J. R. Divine 3" long piece for archives 4" long piece for leaching

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
35	VRM	Brass valve in PVC pipe from high solids manifold line to radiation monitor--sampled from mezzanine level of fuel handling building	>100,000 c/m 45 mRem/hr 25 mR/hr at contact	Opened brass valve and found very coarse gravel-like particles plugging the valve and 1" PVC line--removed particles and bagged--discarded valve and PVC pipe which contained little activity
36	RS-A and B	1 7/8" dia. SS line to reactor sump-pump--sampled at bottom of reactor sump chase--horizontal section before going to filter and pump	1,000 at end of pipe	A. 8" long piece cut into (3) 2 1/2" long pieces for leaching B. 8" long piece sent to J. R. Divine
37	RSE	Black iron elbow connecting 1 7/8" SS pipe to line going to filter and pump of reactor sump--sampled at bottom of reactor sump chase just downstream from RS-A and B	30,000 at end of pipe	Cut off black iron elbow from 4" long piece of SS 1 1/2" dia. pipe--saved pipe and labeled RSE-PIPE
38	WGPT	2 1/2" dia. hole saw plug from 1/2" thick carbon steel waste gas pressurizer tank (large steel tank)--sampled from mezzanine level of fuel handling building where tank stored	<1,000 dpm on inside surface of plug	Leached for radiochemistry
39	RAVD	8 1/2" x 12" section of 1/16" thick galvanized iron reactor air vent duct--sampled from mezzanine level of fuel handling building where ducts stored	<1,000 dpm on inside surface of duct	Cut up for leaching
40	SHFBS-A	1" dia. SS tube from cluster used for storing superheater fuel elements--sampled from bundle No. 1 in fuel storage basin	3,000	Cut into three pieces 3-4" long for leaching
41	SHFBS-B	Same as No. 40	3,000	Sent to J. R. Divine
42	SHFBS-C	Same as No. 40, except taken from bundle No. 2	15,000	Cut into six pieces 2-3" long for leaching
43	MSL-A	2 5/8" dia. hole saw plug from main steam line--6' long piece stored in fuel storage basin	70,000	Counted directly on Ge(Li) and then leached for radiochemistry
44	MSL-B	Same as No. 43	70,000	Sent to J. R. Divine
45	FRSB-A	Piece of SS fuel rack from fuel storage basin--top piece 2" dia. x 5" long piece from east end of rack	250,000	Cut into two 2" long pieces for leaching

Sample Number	ID Number	Sample Description	Exterior Surface Activity (GM-d/m)	Disposition
46	FRSB-B	Same as No. 45	300,000	Sent to J. R. Divine
47	FRSB-C	Same as No. 45, except cut from west end of rack	45,000	Cut into two 2" long pieces for leaching
48	FRSB-D	Same as No. 47	80,000	Sent to J. R. Divine
49	FTC	Piece of fuel transfer chute--4" x 5" SS piece cut from chute near joining ear--sampled from fuel storage basin	15,000	Cut in half - half sent to J. R. Divine - cut other half in two 2" long pieces for leaching
50	FTTR	Fuel transfer tube roller wheel removed from fuel transfer tube reactor sampled from fuel storage basin	0.5 mR/hr 75 mR/hr	Not cut--leach as is
51	FSBTT	2" dia. x 8" long piece of SS cut from fuel storage basin transfer tube for fuel elements--sampled from fuel storage basin	15,000	Cut in half - half sent to J. R. Divine - other half cut into two 2" long pieces for leaching
52	SCDL	1 3/4" dia. steel line draining main steam line of condensate when reactor was down --colleced from storage drum in cage on basement floor of turbine building--some torch cutting on piece	10,000	Cut off "hot" 4" and then cut that into two 2" long pieces for leaching

B.2 HUMBOLDT BAY SAMPLE INVENTORY

Sample Inventory for Humboldt Bay Generating Plant Unit #3

During the period July 13 through July 22, 1981, a sampling program was conducted at the nuclear unit of Pacific Gas and Electric Company's (PG&E) Humboldt Bay Generating Station located near Eureka, California.

An extensive study was conducted at this site since PG&E is currently in the process of deciding the future status of the Humboldt nuclear unit which has been non-operational for approximately six years. The staff at the nuclear station were extremely helpful in delineating appropriate sampling locations and items for sampling. The assistance received from the health physics personnel also expedited the field work and was greatly appreciated.

Samples obtained during the onsite work included twelve soil core samples from areas surrounding the generating station both outside the perimeter fence (seven) and inside the perimeter fence (five), four sediment cores from the intake (two) and discharge canals (two), twenty-seven concrete cores from numerous locations through the generating station, and twenty-six samples from various reactor systems, including scrapings from surfaces, pieces of metal from several reactor systems, and concrete from the biological shield.

SAMPLE IDENTIFICATION AND INVENTORY

Samples were designated by a letter code detailing the origin or sample type as follows:

HBCC -- Humboldt Bay concrete core
HBRS -- Humboldt Bay reactor system
HBSG -- Humboldt Bay soil grab
HBSC-0 -- Humboldt Bay soil core from areas inside the perimeter fence
HBSC -- Humboldt Bay soil core from areas external to the perimeter fence
HBSed -- Humboldt Bay cooling water canal sediment

Humboldt Bay Concrete Cores (HBCC-)

Twenty-seven concrete cores were taken using a concrete coring system shown in Figure 1 which uses an industrial diamond bit and water as a cooling and lubricating medium. The cores are nominally four inches in diameter and vary in depth from approximately 5 cm to over 20 cm. Below is a listing of coring locations in the Humboldt Bay Generating Station.

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Core	Core Location
HBCC-1	4" thick concrete pad used for sandblasting an old turbine.
HBCC-2	Hot Machine Shop. Near floor drain in curbed drain area between railroad tracks.
HBCC-3	Hot Machine Shop. Painted floor between lathe and steel slab floor.
HBCC-4	Rad-waste Tank Area. Taken along (over) crack in the concrete--next to concentrator drip tank.
HBCC-5	Rad-waste Tank Area. Taken 2-3 feet away from HBCC-4 and having no crack.
HBCC-6	Rad-Waste Building. Under rad-waste sampling station.
HBCC-7	Rad-Waste Building. Inside the step-off pad to the concentrator feed pump area.
HBCC-8	Reactor Building. -66' level outside step-off pad at the bottom of the man-lift.
HBCC-9	Reactor Building. -66' level inside step-off pad and inside exclusion fence.
HBCC-10	Reactor Building. -66' level directly under pressure vessel.
HBCC-11	Reactor Building. -34' level in the center of the concrete floor area. Near suppression chamber hatch cover plate.
HBCC-12	Reactor Building. -24' level in the center of the concrete floor area.
HBCC-13	Concrete roof over CWT and RDT vault.
HBCC-14	Asphalt. Taken through asphalt in area where spill had occurred and had been painted over with sealer.
HBCC-15	Condensate Demineralizer Room. Core taken in hallway near drain.
HBCC-16	Condensate Demineralizer Room. Core taken over crack in floor in the regeneration room.
HBCC-17	Condensate Demineralizer Room. Core taken over solid floor near HBCC-16 in regeneration room.

HBCC-18	Condensate Pump Room. Core taken in the middle of the concrete floor area.
HBCC-19	Turbine Building. Core taken directly under the turbine near floor drain.
HBCC-20	Turbine Building. Core taken in the pipe tunnel between old admiralty brass heat exchangers.
HBCC-21	Turbine Building. Next to the condenser at the bottom of first flight of stairs near condensers.
HBCC-22	In Yard. Core taken on concrete pad near stack and lid to the offgas filter system.
HBCC-23	In Yard. Core taken on the north side of condensate storage tank.
HBCC-24	Air Ejector Room. Core taken on the northeast end of the room near floor drain.
HBCC-25	Reactor Building Refuel Level. Core taken between railroad tracks by the large double doors on the northeast corner of the building.
HBCC-26	Background sample. Clean core taken on pumping platform at water intake canal.
HBCC-27	Access Control. Core taken beyond step-off area in the aisle on northwest end of the room.

Humboldt Bay Reactor Systems Samples (HBRS-)

Reactor system samples were of several types. These included scrapings of corrosion films from the inside surfaces of several reactor systems, water samples from the spent fuel storage pool and steam suppresion chamber, and pieces of hardware when such was available. The scrapings were taken using a paint scraper and a sample obtained by scraping a specified surface area. The samples are listed below by number and the origin and type of sample specified.

Sample	Location Description
HBRS-1	Spent Fuel Storage Pool. Scrappings from the side of the pool basin. Scrapings from 8 x 25 cm area. The pool had been previously higher than it was at the time of our sampling. Thus, there was a thin layer of material on the wall deposited during the period of higher water.

HBRS-2	Spent Fuel Pool. A section of cable from which the feed water sparger was suspended for storage in the spent fuel pool.
HBRS-3	Spent Fuel Storage Pool. 1 liter of water was obtained from the storage pool using sampling line available on the refuel level.
HBRS-4	Steam Suppression Chamber. 1 liter of water was sampled.
HBRS-5	Steam Suppression Chamber. A smear of approximately 900 cm^2 was obtained from the suppression chamber wall just above the ambient water level in the chamber.
HBRS-6	Steam Relief Valve. An area of approximately 65 cm^2 was scraped on the inside surface of the relief valve.
HBRS-7	Main Steam Condenser. A smear of the sludge on a horizontal surface inside the condenser was obtained. Smear area was approximately 25 cm^2.
HBRS-8	Main Steam Condenser. An area of approximately 25 cm^2 was scraped on a vertical plate inside the condenser.
HBRS-9	Emergency Condenser. An area of approximately 400 cm^2 was scraped on the inside of the condenser.
HBRS-10	Gland Seal Exhauster-Condensate Outlet. Scrapings were taken from the orifice plate which had an 8" O.D. and 5" I.D. A red oxide film was present on both sides of the plate.
HBRS-11	Condensate Demineralizer System. Scrapings were taken from the inside of the #1 condensate-demineralizer strainer.
HBRS-12	Offgas HEPA Filter Assembly. Scrapings were taken from a horizontal plate inside the offgas vault from a 25 cm^2 area.
HBRS-13	Offgas HEPA Filter Assembly. Sludge was obtained from a small pool of standing water inside the filter assembly.
HBRS-14	Air Ejector. A Y-strainer was obtained from the air ejector system.
HBRS-15	Air Ejector. Scrapings were obtained from the inside of a 1" diameter pipe. An area approximately 2.5 cm long was scraped from the Y-strainer system.

HBRS-16	Control Rod Drive Hydraulic Filters. Scrapings were taken from the inside surface of an inlet filter. Condensate was used as the hydraulic fluid for the control rod drive system 25 cm^2 area was scraped.
HBRS-17	Control Rod Drive Hydraulic Filters. Particulate matter (crud) was tapped from the CUNO filter assembly for the inlet hydraulic fluid (condensate).
HBRS-18	Reactor Feedwater System. A sparger clamp bracket was obtained which had been in the feedwater system.
HBRS-19	Reactor Feedwater System. A short section of reactor feedwater line was obtained from a previously discarded section of line in storage on the refuel level.
HBRS-20	Sump Drains. Sludge collected from the sump drains at the -66' level was sampled. The sludge was being stored in barrels.
HBRS-21	Sump Drains. Another aliquot of sludge was sampled from a second storage barrel.
HBRS-22	Biological Shield. Concrete from the edge of the biological shield nearest to the reactor pressure vessel just above the -66' level was obtained using a rotary hammer.
HBRS-23	Biological Shield. Concrete from the edge of the biological shield away from the pressure vessel just above the -66' level was obtained using a rotary hammer.
HBRS-24	Biological Shield. Concrete from the upper corner of one portion of the removable upper shield ring was sampled using a rotary hammer. Sampling was conducted at the refuel floor level.
HBRS-25	Biological Shield. Concrete from the lower corner of one portion of the removable upper shield ring was sampled using a rotary hammer. Sampling was done at the refuel floor level.
HBRS-26	Regenerative Heat Exchanger. A short elbow was cut from one of the old regenerative heat exchangers.

Humboldt Bay Soil Grabs (HBSG)

Three samples of surficial soil were obtained as grab samples in areas of known contamination from previous work or spills in the yard area of the plant. These are detailed below:

Sample	Description
HBSG-01	Soil taken near the exhaust gas stack on the southeast side. Soil taken to approximately 2.5 cm depth.
HBSG-03	Soil taken between the sandblasting pad and the perimeter fence at a hot spot reading 500 c/m on GM. Surface soil to approximately 2.5 cm taken.
HBSG-03A	A second hot spot near the sandblasting pad. Soil taken to 2.5 cm.

Humboldt Bay Soil Cores Inside the Perimeter Fence (HBSC-0)

Two soil cores were obtained inside the perimeter fence, one in an area where a spill had occurred in the past and the second on a hillside which received no known contamination. The two samples and their origins are listed below:

Sample	Description
HBSC-01	A 50 cm soil core was obtained on the north side of the condensate storage tank. A surface layer of aggregate (2.5 cm) was set aside and the underlying soil sampled at approximately 5 cm intervals.
HBSC-02	A 35 cm soil core was obtained in the grassy median between the roadway to the rad-waste tank area and the roadway up the hill to the storage area. It was sectioned into approximately 5 cm intervals.

Humboldt Bay Soil Cores Outside of Exclusion Area (HBSC-)

Seven soil cores were taken surrounding the reactor. Sampling locations are shown in Figure 2.2. These sites were shown after consultation with plant personnel and should reflect a background location and the potential fallout plume during the predominant wind regimes at the site.

Sample	Description
HBSC-1-1	A 4 cm soil core in 1 cm intervals obtained directly "called north" of the reactor.
HBSC-1-2	A 4 cm soil core in "called northeast" direction from plant. The core was segmented into 1 cm intervals.
HBSC-1-3	A 4 cm soil core "called east" from the reactor was segmented into 1 cm intervals.
HBSC-1-4	A 4 cm soil core was obtained and segmented into 1 cm intervals in the "called south southwest" direction from the reactor.
HBSC-1-5	A 4 cm soil core in the "called west southwest" direction from the reactor was segmented into 1 cm intervals.
HBSC-1-6	A 4 cm soil core, segmented into 1 cm intervals, was obtained in the direction "called west" from the reactor.
HBSC-1-7	A 4 cm soil core, segmented in 1 cm intervals, was obtained in the direction "called northwest" from the reactor.

Humboldt Bay Sediments (HB-Sed.)

A total of four sediment samples were obtained from the intake and outlet sides of the cooling water canal using a plastic corer or by grab sampling. The locations and descriptions are given below:

Sample	Description
HB-Sed. 1	A 25 cm deep core was taken from the intake water canal near the entrance gate to the plant parking lot. The core was segmented at approximate 6 cm intervals.
HB-Sed. 2	A 25 cm deep core was taken approximately 45 meters downstream from the discharge inlet. The core was subsequently sectioned in 6 cm intervals.

HB-Sed. 3	A grab samples of the surficial 10 cm of sediment was collected from the intake platform.
HB-Sed. 4	A 25 cm deep core was taken in the discharge canal approximately 20 meters from the outlet end. It was sectioned into four segments of approximately 6 cm each.

B.3 SAMPLE INVENTORY FOR DRESDEN NUCLEAR POWER STATION UNIT ONE

SAMPLE INVENTORY FOR DRESDEN NUCLEAR POWER STATION UNIT ONE

During the period August 2 through August 11, 1982, a sampling program was conducted at Commonwealth Edison's Dresden Nuclear Power Station Unit No. 1.

An extensive sampling program was conducted at this site since the unit has not been operational since October, 1978, and the future status of the unit is uncertain. Commonwealth Edison is considering whether to update to current safety standards or decommission the unit. The staff at the nuclear station was extremely helpful in selecting appropriate sampling locations and hardware for sampling.

Commonwealth Edison is currently planning a decontamination effort in the unit. Therefore, it was necessary to maintain operational system integrity after sampling. Consequently, the sampling hardware was limited to scraping interior surfaces of opened systems.

Samples obtained during the onsite work included five surficial soil core samples from areas associated with Unit No. 1, sediment cores from the intake and discharge canals, nine concrete cores at selected locations through the nuclear unit, and thirteen scraping samples from the various reactor systems.

SAMPLE IDENTIFICATION AND INVENTORY

Samples were designated by a letter code detailing the origin or sample type as follows:

D-CC - Dresden, Concrete Core

D-SC - Dresden, Scraping from Reactor Systems

D-SS - Dresden, Surficial Soil Sample

D-Sed - Dresden, Sediment Core

DRESDEN UNIT No. 1 CONCRETE CORES

Nine concrete cores were taken using a coring system employing an industrial diamond bit and recirculated water as a cooling and lubricating medium. The cores are normally four inches (10 cm) in diameter and range in depth from

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5 cm to approximately 20 cm. The table below gives a detailed listing of the coring locations and dose rate data associated with the surface face of each core.

PNL - ID	Dose Rate		Location
	γ	β	
DCC-1	13 mR/hr	12 mRad/hr	Dow Chemical Spill Area - 529' Containment
DCC-2	--	300	Secondary Steam Generator Room "B" - 529' Containment
DCC-3	--	<1	Hall in Front of Accumulator Room - 488' Containment
DCC-4	20	420	Sub-pile Room - 488' Containment
DCC-5	--	<1	Clean Makeup Demineralizer Room - 515' Turbine Bldg.
DCC-6	--	22	Condensate Pump Room Pit Turbine Bldg.
DCC-7	4	230	Unloading Heat Exchanger Room - 517' Containment
DCC-8A	2	32	Rad Waste Bldg. Basement Top of Core - 490'
DCC-8B	--	--	Rad Waste Bldg. Basement Bottom of Core - 490'
DCC-9	--	300	Rad Waste Bldg. Basement North - 490' End Drainage Trough

DRESDEN REACTOR SYSTEM SAMPLES

The reactor system samples at Dresden were all scrapings of corrosion films associated with steel piping or plate except for the floor drain from Instrument Room A. These samples were obtained by scraping the interior surface of the various systems as made available by plant operations personnel. The scrapings were taken using a paint scraper and specified surface areas were scraped. The samples are listed below by number; the location and dose rate associated with the sample are also indicated.

	Dose Rate			
PNL - ID	γ	β	Location	
D-SC-10	13 mR/hr	30 mRad/hr	Fuel Storage Pool Wall	517'
D-SC-11	4	9	Fuel Pool Water Circulation Pump	517'
D-SC-12	14	120	Fuel Pool Metal Ladder	517'
D-SC-13	--	--	Inside Sphere Wall	529'
D-SC-14	--	--	Containment Air Vent Inside Sphere	529'
D-SC-15	8	98	Floor Drain - Instrument Room A	502'
D-SC-16	4	12	Fuel Transfer Chute Bottom of Fuel Canal	--
D-SC-17	4	9	Fuel Canal Scrapings	559'
D-SC-18	400	1900	Reactor Steam Vent Line	584'
D-SC-19	--	--	Steam Relief Valve	--
D-SC-20	--	--	Low Pressure Inlet to Turbine	--
D-SC-21	--	--	High Pressure Outlet from Turbine	--
D-SC-22	--	--	Steam Line Piping	--

DRESDEN SOIL AND SEDIMENT SAMPLES

Seven samples (five soils, two sediments) were obtained from areas associated with Unit One activities. The samples are listed below with depths and sampling locations. The sediment cores were obtained from the canal bank using a plastic corer attached to a long hand held aluminum extension.

PNL I.D.	Depths (cm)	Location
D-SS-23	0-7.5	North Side of Containment Sphere near Equipment Transfer Hatch
D-SS-24	0-7.5	East Side of Containment Sphere near Equipment Storage Area

D-SS-25	0-15	Rad Waste Area - in front of building doorway
D-SS-25B	15-30	Rad Waste Area - in front of building doorway
D-SS-26	0-7.5	Rad Waste Storage Tank Area
D-SS-27	0-7.5	South Side of Containment Sphere near refueling building
D-Sed-28	0-20	Inlet Canal Sediments
D-Sed-29	0-20	Outlet Canal Sediments

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B.4 SAMPLE INVENTORY FOR INDIAN POINT STATION, UNIT ONE

Sample Inventory for Indian Point Station, Unit One

The Indian Point Station, Unit 1 was sampled over the period of May 18, 1982 to May 28, 1982. The samples obtained were concrete cores, hardware, scrapings from various surfaces and piping, and concrete chips from the bio-shield under the reactor.

The sample designations are as follows:
IPCC: Indian Point Station, Unit One concrete core sample
IPH: Indian Point Station, Unit One hardware sample.

A list of the concrete cores collected along with a brief description of the sample location follows:

IPCC-1 5/21/82	Concrete core collected on floor near primary blowdown coolers on the 53' level in the Vessel Containment building.
IPCC-2 5/21/82	Concrete core collected on floor near south end of Nuclear Boiler No. 14 on the 70' level in the Vessel Containment building.
IPCC-3 5/21/82	Concrete core collected on floor on NE side of No. 12E decay heat pump on the 33' level in the Vessel Containment building.
IPCC-4 5/21/82	Concrete core collected on floor walkway beneath the transfer tube on the 15' level in the Vessel Containment building.
IPCC-5 5/21/82	Concrete core collected on east side of pit floor on 5' level in the Vessel Containment building (near IPCC-7).
IPCC-6 5/21/82	Concrete core collected on north side of pit floor under the reactor on the 5' level in the Vessel Containment building (near IPCC-8 and -9).

IPCC-7 5/24/82	Concrete core collected on east side of pit floor under the reactor on the 5' level in the Vessel Containment building (near IPCC-5).
IPCC-8 5/24/82	Concrete core collected on north side of pit floor on 5' level in the Vessel Containment building (near IPCC-6 and 9).
IPCC-9 5/24/82	Concrete core collected north side of pit floor under the reactor on the 5' level in the Vessel Containment building (near IPCC-6 and 8).
IPCC-10 5/24/82	Concrete core collected on floor in secondary valve gallery on 33' level in the Nuclear Service building.
IPCC-11 5/24/82	Concrete core collected on floor in contaminated waste drain equipment room on 33' level in the Nuclear Services building.
IPCC-12 5/24/82	Concrete core collected on floor by No. 10 drain tank on 33' level of the Chemical Services building.
IPCC-13 5/24/82	Concrete core collected on floor by No. 12 sea water and primary makeup pumps on the 14' level in the Chemical Services building.
IPCC-14 5/24/82	Concrete core collected on floor in the secondary blowdown purification pump room on the 33' level of the Chemical Services building.
IPCC-15 5/24/82	Concrete core collected on floor by clean water transfer pump on the 14' level of the Chemical Services building.

Table C1. Survey Data of Concrete Core Samples

ID	Date	Bldg.	Elev.	Smear (dpm/100 cm^2)	Survey Date of Sample		
					β mrad/hr	γ mR/hr @ contact	γ mR/hr @ 1'
IPCC-1	5/21/82	VC	53'	20,000	20	<1	<0.1
IPCC-2	5/21/82	VC	70'	40,000	8	<1	<0.1
IPCC-3	5/21/82	VC	33'	50,000	44	<1	<0.1
IPCC-4	5/21/82	VC	15'	2,000	80	1	0.1
IPCC-5	5/21/82	VC	5'	200,000	6400	100	2
IPCC-6	5/21 2	VC	5'	100,000	1200	25	1
IPCC-7	5/24/82	VC	5'	10,000	240	14	0.5
IPPC-8	5/24/82	VC	5'	5,000	200	2	0.2
IPPC-9	5/24/82	VC	5'	200,000	5600	50	1
IPCC-10	5/24/82	NSB	33'	1,000	ND	<1	<0.1
IPCC-11	5/24/82	NSB	33'	3,500	ND	<1	<0.1
IPCC-12	5/25/82	CSB	33'	1,000	160	0.5	<0.1
IPCC-13	5/25/82	CSB	14'	1,000	ND	<1	<0.1
IPCC-14	5/25/82	CSB	33'	1,500	80	1	<0.1
IPCC-15	5/25/82	CSB	14'	50,000	2400	21	1

VC = Vessel Containment building
NSB = Nuclear Services building
CSB = Chemical Services building
ND = Not detectable

A list of the hardware samples collected along with a description of the sample location follows:

IPH-1 5/25/82	Sample of the steel support plate to the reactor pressure vessel obtained from the 33' level of the Vessel Containment building.
IPH-2 5/25/82	Concrete chips from the inside surface of the bioshield under the reactor vessel obtained on the 33' level of the Vessel Containment building.
IPH-3 5/25/82	Sample of seal water line from the 12 east off-heat decay pump on the 33' level of the Vessel Containment building.
IPH-4 5/25/82	Sample of boric acid residue from the seal water line - 12 east off-heat decay pump on the 33' level of the Vessel Containment building.
IPH-5 5/26/82	Scrapings from the annulus (containment sphere) wall (929 cm^2) collected from the 108' level of the Vessel Containment building.
IPH-6 5/26/82	Scrapings from the annulus (containment sphere) wall (929 cm^2) collected from the 70' level of the Vessel Containment building.
IPH-7 5/26/82	Scrapings from the annulus (containment sphere) wall (929 cm^2) collected from the 53' level of the Vessel Containment building.
IPH-8 5/26/82	Scrapings from the annulus (containments sphere) wall (929 cm^2) collected from the 33' level of the Vessel Containment building.
IPH-9 5/26/82	Scrapings from the annulus (containment sphere) floor near the fuel transfer chute on the 33' level of the Vessel Containment building.

IPH-10 5/26/82	Pipe from No. 14 primary coolant pump collected on the 88' level of the Vessel Containment building.
IPH-11 5/26/82	Tube from the radiation monitoring tubing for the secondary blowdown in cell No. II. Collected from the 53' level of the Nuclear Service building (0.5 in ID X 40 in, stainless steel tube).
IPH-12 5/26/82	Pipe going to reactor valve SA250 from cell No. III on the 53' level of the Nuclear Service building (0.5 in ID X 4.5 in with 0.5 in elbow, carbon steel).
IPH-13 5/26/82	Tubing to loop No. 4 of the boronometer in cell No. III on the 53' level of the Nuclear Service building (0.25 in ID X 10.5 in).
IPH-14 5/26/82	Inner seal water sampling line from cell No. III on the 53' level of the Nuclear Service building.
IPH-15 5/26/82	Scrapings from valve SD253 in the secondary valve gallery on the 33' level of the Nuclear Service building (40.5 cm^2).
IPH-16 5/27/82	Floor scrapings of resin in the resin sluice tank room on the 14' level of the Chemical Service building.
IPH-17 5/27/82	Scrapings from the vertical floor pipe in the resin sluice tank room on the 14' level of the chemical services building (81.1 cm^2).
IPH-18 5/27/82	Scrapings from line No. 13, purification transfer pumps, located on the 14' level of the Chemical Service building (122 cm^2).
IPH-19 5/27/82	Waste outlet filter scrapings from the exterior surface of the filter housing line located on the 33' level of the Chemical Services building (155 cm^2).
IPH-20	Stainless steel elbow from No. 10 clean water transfer pump on the 14' level of the Chemical Services building (2.5 in ID X 8

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in long with 1.5 in ID X 4 in reducing elbow).

IPH-21 5/27/82	Scrapings from the wall of the fuel storage pool (3" X 3") on the 70' level of the Fuel Service building (58 cm^2).
IPH-22 5/28/82	Scrapings from boiler No. 14 stainless steel manway located on the 70' level of the vessel containment building (103 cm^2).
IPH-23	Scrapings from boiler No. 14, manway diaphragm, stainless steel, located on the 70' level of the Vessel Containment building (103 cm^2).
IPH-24	Floor tile from a hot spot in the Vessel Containment building (522 cm^2).

TABLE C2. Survey Data for the Hardware Samples

ID	Date	Bldg.	Elev.	Smear (dpm/100 cm^2)	Survey Data of Sample		
					β mrad/hr	γ mR/hr @ contact	γ mR/hr @ 1'
IPH-1	5/25/82	VC	33'	1,500	ND	<1	<0.1
IPH-2	5/25/82	VC	33'	NA	20	<1	<0.1
IPH-3	5/25/82	VC	33'	40,000	20	0.5	<0.1
IPH-4	5/25/82	VC	33'	NA	320	10	1
IPH-5	5/26/82	VC	88'	NA	ND	<1	<0.1
IPH-6	5/26/82	VC	70'	NA	ND	<1	<0.1
IPH-7	5/26/82	VC	52'	NA	ND	<1	<0.1
IPH-8	5/26/82	VC	33'	NA	ND	<1	<0.1
IPH-9	5/26/82	VC	33'	NA	20	<1	<0.1
IPH-10	5/26/82	VC	88'	3,700	ND	<1	<0.1
IPH-11	5/26/82	NSB	53'	1,000	4	<1	<0.1
IPH-12	5/26/82	NSB	53'	36,000	40	3	0.3
IPH-13	5/26/82	NSB	53'	7,400	ND	<1	<0.1
IPH-14	5/26/82	NSB	53'	1,800	14	<1	<0.1

TABLE C2 (continued). Survey Data for the Hardware Samples

					Survey Data of Sample		
ID	Date	Bldg.	Elev.	Smear (dpm/100 cm^2)	β mrad/hr	γ mR/hr @ contact	γ mR/hr @ 1'
IPH-15	5/26/82	NSB	33'	NA	ND	<1	<0.1
IPH-16	5/27/82	CSB	14'	NA	520	50	2
IPH-17	5/27/82	CSB	14'	NA	ND	<1	<0.1
IPH-18	5/27/82	CSB	14'	NA	400	5	0.5
IPH-19	5/27/82	CSB	33'	NA	120	3	0.3
IPH-20	5/27/82	CSB	14'	NA	100	2	0.2
IPH-21	5/27/82	FSB	70'	NA			
IPH-22	5/28/82	VC	70'	NA	280	75	10
IPH-23	5/28/82	VC	70'	NA	70	2	0.2
IPH-24		VC					

VC = Vessel Containment building
NSB = Nuclear Service building
CSB = Chemical Service building
FSB = Fuel Service building
NA = Not analyzed
ND = Not detected

B.5 MONTICELLO SAMPLE INVENTORY

MONTICELLO STATION SAMPLE INVENTORY AND DISPOSITION
4-9 May 1981 Sampling

Hardware

Sample Number	Sample Description	Dose Rate (mR/hr) or GM counts/minute of analyzed sample	Sample Disposition
	Condensate demin. filter assembly		
M-100-6	Spring seats, 50 each, 0.5 mR/hr	1100 cpm (0.5 mR/hr)	Analyzed 17 each
M-100-5	Springs, 7 each, <0.2 mR/hr	1100 cpm (<0.2 mR/hr)	Analyzed 2 each
M-100-3	Extension nuts, 12 each, 0.2 mR/hr	1000 cpm (0.2 mR/hr)	Analyzed 4 each
M-100-4	Guide rods, 3 each, <0.5 mR/hr	400 cpm (<0.5 mR/hr)	Analyzed 1 each
	Feedwater heater #14A impinger plate		
M-100-2	10 to 15 cm long pieces of 1.9 cm tubing cut off plate, 3 each	2000 cpm (<1 mR/hr)	Analyzed 1 piece
M-100-1	Two corners of the plate (8 cm x 8 cm) cut off, 2 each	8000 cpm (1.5 mR/hr)	Analyzed 1 piece
	Silver plated metal O-ring from reactor pressure vessel head		
M-100-7	About 122 cm long piece cut into ~10 cm lengths	9500 cpm	Analyzed 88 cm section

Concrete Cores (10 cm diameter)

	Building	Elevation	Location	Depth (cm)
M-CC-1	Reactor Bldg	1001'2" elev	Tool decon area	7.6
M-CC-3	Reactor Bldg	1001'2" elev	Tool decon area	5.7
M-CC-8	Reactor Bldg	962'6" elev	Reactor water cleanup pump room	12.7
M-CC-9	Reactor Bldg	985'6" elev	Reactor radwaste pump room	12.7
M-CC-7	Reactor Bldg	1027'8" elev	Refuel floor	12.7
M-CC-10	Reactor Bldg	985'6" elev	Snubber mainten. shop	12.7
M-CC-4	Radwaste Bldg	935'0" elev	Shipping area	10.8
M-CC-5	Radwaste Bldg	935'0" elev	Compactor area	10.0
M-CC-2	Turbine Bldg	Operating floor	Near feedwater heater	5.0
M-CC-6	Turbine Bldg	Operating floor	Condensate demin. work area	5.0

Soils (0.1 m^2 area x 3 cm deep)

M-SS-2	West of Reactor Building	<200 cpm	Analyzed 200 g of each soil
M-SS-1	Northwest of Reactor Building	<200 cpm	
M-SS-4	East of Reactor Building	<200 cpm	
M-SS-3	South of Reactor Building	<200 cpm	

MONTICELLO STATION SAMPLE INVENTORY AND DISPOSITION
19-20 November 1981 Sampling

Piping and Hardware

Sample Number	Sample Description	Dose Rate (mR/hr) or GM cpm	Sample Disposition
M-101	9.2 cm diameter (OD) reactor water cleanup line (inlet to demins.), 11 years in service, 985' pump room, 4 pieces cut 15 to 30 cm long	100 mR/hr	Analyzed one 17 cm long piece
M-102	2.5 cm diameter steam drain piping, 6 years in service, constant flowing, 3 pieces 15 cm long each	2000 cpm	Analyzed one 8.3 cm long piece
M-103	25 cm diameter condensate piping, 11 years in service, 2 pieces about 15 cm long each	--	Analyzed one 10 cm long piece
M-104-L M-104-V	1.3 cm diameter instrument lines, condensate, 11 years in service, little flow several feet cut into 20 cm long sections, plus short pieces of misc. pipe and valve M-104V = Valve M-104L = 1/2" pipe	200 cpm	Analyzed valves and 58 cm long piece of pipe
M-105	Turbine diaphram scrapings, scraped 2 diaphrams, 160 cm^2 area each and combined scrapings, turbines and diaphram 11 years in service	1.5 mR/hr	
M-106	Sample of sand blasting grit used for blasting turbine and turbine diaphrams, 0.9 kg	500 cpm	Analyzed 100 gm aliquot
M-107A M-107B M-107C	Condensate storage water valve, 4 months in service, valve normally in closed position M-107A = valve M-107B = valve M-107C = piping	< 1 mR/hr	Analyzed each valve and all piping
M-108	Screen from filter at exit of condensate demin/filt	< 1 mR/hr	Analzyed whole filt

M-109	Pump from rad-waste drain from off-gas drains	< 1 mR/hr	Analyzed contaminated internal surface of pump

B.6 TURKEY POINT SAMPLE INVENTORY

SAMPLE INVENTORY FOR TURKEY POINT

During the period October 12 through October 22, 1981, a sampling program was conducted at Units 3 and 4 of the Florida Power and Light, Turkey Point Generating Station located near Homestead, Florida.

Unit 3 was in the midst of a steam generator replacement outage and Unit 4 began an outage for the same purpose during the time of our sampling visit. Since these were both operating units, our sampling was limited to samples of opportunity, i.e. components removed from service during the outages. The staff at the nuclear station were very helpful in delineating appropriate sampling locations and equipment available for subsampling.

Samples obtained during the onsite work included eleven shallow soil core samples from areas inside the perimeter of the security fence and inside the radiation controlled area. Also, one sediment sample was obtained from the cooling water discharge canal. Sixteen concrete cores were obtained from areas throughout the rad waste building (six), the auxiliary building (seven), and containment (three). Ten samples from reactor systems were obtained including pieces of hardware, scrapings, alumina grit, and piping.

SAMPLE IDENTIFICATION AND INVENTORY

Samples were designated by a letter code detailing the origin of the type of sample as follows:

TPCC	Turkey Point Concrete Core
TP-H	Turkey Point Hardware
TP	Turkey Point Soil
TP-Sed	Turkey Point Sediment from the Cooling Canal

TURKEY POINT CONCRETE CORES

Sixteen concrete cores were obtained using a concrete coring system containing an industrial diamond bit, and which employs water as a cooling and lubricating medium. The cores were nominally four inches in diameter and varied in depth from 5 to 10 cm. Below is a listing of the coring locations and the surface activity as measured by GM meter or dose rate meter.

Core	Location	Surface Activity or Dose Rate
TPCC-1	Decon area floor, Rad. Waste Bldg.	<200 cpm
TPCC-2	South filling room floor, Rad. Waste Bldg.	<200 cpm
TPCC-3	South filling room gutter, Rad. Waste Bldg.	4,500 cpm
TPCC-4	Pipe and Valve Room (above pipe tunnel), Rad. Waste Bldg.	1,500 cpm
TPCC-5	Pipe tunnel, Rad. Waste Bldg.	1,500 cpm
TPCC-6	Waste evaporator room, Rad. Waste Bldg.	300 cpm
TPCC-7	Boric Acid Evapor. room (No. 4), Aux. Bldg.	3,500 cpm
TPCC-8	Gas stripper room, Aux. Bldg.	600 cpm
TPCC-9	Concentrate tank holding room, Aux. Bldg.	8,000 cpm
TPCC-10	Hallway (next to drain sump), Aux. Bldg.	1,000 cpm
TPCC-11	Spent fuel heat exchanger room, Aux. Bldg.	2,000 cpm
TPCC-12	Regenerative heat removal (RHR) room, Aux. Bldg.	4,500 cpm
TPCC-13	Fuel storage basin walkway, Aux. Bldg.	800 cpm
TPCC-14	Reactor circulation pump floor, Containment Bldg.	300 cpm
TPCC-15	Reactor sump floor (Unit 4)	6 mR/hr
TPCC-16	Reactor sump floor (Unit 4)	2 mR/hr

TURKEY POINT REACTOR SAMPLES

Reactor samples were of several types. They included scrapings of corrosion films from surfaces of several reactor system components, pieces of plate and pipe cut from equipment removed during the outage, and a sample of decon grit used on the steam generator. The scraping samples were taken using a paint scraper over a specified area. The samples are listed below, and the table contains pertinent information regarding source, area represented or weight, and radiation level as measured by a GM meter or dose rate meter.

Sample	Source	Area (cm^2)	Wt. (g)	Activity
TP-H-1	Steam Dryer - 2 pieces of pipe and plate	580	765	<200 cpm
TP-H-2	Steam Dryer - 2 bars from sparger pipes	430	2845	<200 cpm
TP-H-3	Scraping from main stream line	290	10.6	<200 cpm
TP-H-4	Scraping from secondary feedwater line	290	5.8	<200 cpm
TP-H-5	Scraping from control rod ventilation ducting	290	-	<200 cpm
TP-H-6	Secondary blowdown line from steam generator	930	3200	<200 cpm
TP-H-7	Alumina decon grit from steam generator		29.2	190 mR/hr
TP-H-8	Scraping from "C" steam generator manway	206	-	13 mR/hr
TP-H-9	Scraping from "C" steam generator bowl top 6"	206		5000 cpm
TP-H-10	Reactor purification pipe - 2" diameter	-	-	1 mR/hr, 8500 cpm

TURKEY POINT SOILS

Ten surficial soil samples were obtained from the vicinity of Units 3 and 4 at Turkey Point. Six of the samples were obtained within the radiation controlled area, i.e. within approximately 50 meters from the nuclear units. Three of the radiation controlled area samples were taken from within a fenced, posted contamination area.

Four surficial soil samples were taken within the security fence at distances ranging from approximately 100 to 500 meters. No detectible radionuclides above fallout background.

Finally one sediment sample was obtained from the cooling effluent canal from the site.

The samples, their identification and locations and the depth intervals sampled are tabulated below.

Sample	Location	Depth Intervals
TP-1	30 yds E. of Unit 3	0-4, 4-8
TP-2	30 yds E. of Unit 4	0-4, 4-8
TP-3	30 yds W. of Unit 4	0-4, 4-8
TP-4	200 yds E. of Unit 3	0-4, 4-8
TP-5	500 yds NE. of Unit 1	0-4, 4-8
TP-6	100 yds W. of Unit 3	0-4, 4-8
TP-7	400 yds N. of Unit 1	0-4, 4-8
TP-8	50 yds SE. of Unit 4 (fenced, posted, controlled area); NW Sector	0-4, 4-8
TP-9	50 yds SE. of Unit 4 (fenced, posted, controlled area); NW corner	0-5
TP-10	50 yds SE. of Unit 4 (fenced, posted, controlled area)	0-5
TP-11	Sediment from discharge canal	0-5

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B.7 RANCHO SECO SAMPLE INVENTORY

During the periods March 20 to 24 and June 7 to 10, 1983 a sampling program was conducted at the Rancho Seco nuclear generating station. Sampling was conducted during an extended outage at the station in which TMI related modifications were being made.

Samples obtained during the sampling included five hardware samples, five scrapings from available surfaces, four surficial soils, four sediments from cooling spray ponds, and two concrete cores.

Sample Identification and Inventory

Samples were designated by a letter code detailing the origin or sample type as follows:

RS-CC- -Rancho Seco, concrete core
RS-Sc- -Rancho Seco, scraping
RS-H- -Rancho Seco, hardware
RS-SS- -Rancho Seco, surficial soil
RS-Sed -Rancho Seco, Sediments

Reactor System Samples

Both pieces of hardware (5) and surficial samplings (5) were obtained at Rancho Seco. These are detailed in the tables below. The scrapings were obtained through the use of a paint scraper over a specified surface area. Hardware samples were obtained from pieces made available as a result of system modifications.

Hardware Samples

PNL Ident.	Dose Rate	Description
RS-H-1	120 mR/hr contact	2" dia. elbow from primary coolant high pressure injection
RS-H-2	150 mR/hr	HEPA filter from upper primary side of steam generator repair tent used during May, 1981 outage

PNL Ident.	Dose Rate	Description
RS-H-3	-	Chevron baffle plates from main steam reheaters (original equipment).
RS-H-4V	70 mR/hr	Valve from letdown cooling system
RS-H-4	200 mR/hr	2 1/2" pipe from letdown cooler

Scraping Samples

PNL Identification	Description
RS-Sc-1	Scraping from secondary side of steam generator OSG-B, near inside top of manway (10 cm x 10 cm area).
RS-Sc-2	Scraping from secondary side of steam generator OSG-A, near inside top manway (10 cm x 10 cm area).
RS-Sc-3	Scraping of boron residue from wall of Turbine Bldg., grade level, east wall (30 cm x 30 cm area).
RS-Sc-4	Scraping of boric acid crystals from wall underneath fuel transfer chute at - 27' level in containment (30 cm x 30 cm area).
RS-Sc-5	Scraping from inside turbine housing (60 cm x 60 area).

Four soil samples and four sediment samples were obtained from the Rancho Seco site and are detailed below in two tables. Soil samples were grab samples of the top layer of soil after removal of a surficial aggregate layer. Sediments were obtained by scraping the sides or bottom of the concrete surface in the retention basin or spray pond.

Surficial Soils

PNL Identification	Area Dose Rate	Description
RS-SS-1	2000 μR/hr	Grab sample obtained from the west side of borated water storage tank. Aggregate was removed from surface and the upper 5-10 cm of soil sampled.
RS-SS-2	110	Grab sample taken on the west side of the demineralized reactor coolant storage tank. Aggregate was removed and the soil below sampled. Top 5-10 cm of soil were removed.
RS-SS-3	140	Grab sample obtained from the east side of the demineralized reactor coolant storage tank. The sampling procedure followed was as the soils above.
RS-SS-4	400	Soil grab sample taken from the east side of the spent fuel storage pool water sampling area.
RS-SS-5	1.2	Grab sample obtained 10' west of Transformer Yard.
RS-SS-6	2.0	Grab sample obtained from the West end of Transformer Yard, 15 feet from station Service Transformer E.
RS-SS-7	2.5	Grab sample taken 5 feet south from the perimeter wall, 60 feet north of Reactor Building.

Sediments

PNL Identification	Description
RS-Sed-1	Sediments removed from the bottom of the South retention basin
RS-Sed-2	Residue scraped from the discharge piping in the South retention basin
RS-Sed-3	Sediments obtained from the South water retention basin
RS-Sed-4	Sediments removed from the spray pond

Concrete Cores

Only two concrete cores were obtained at Rancho Seco. A water-cooled, diamond bit coring system was utilized as at the previous nuclear stations. The cores were approximately 10 cm in diameter and 8-10 cm in depth.

PNL Identification	Description
RS-CC-1	-27' level near "A" sump next to "A" steam generator. 8 cm deep
RS-CC-2	Core approximately 1 meter south of RS-CC-1

APPENDIX C

APPENDIX C

RADIONUCLIDE CONCENTRATIONS AND INVENTORIES IN PIPING, HARDWARE, CONCRETE, AND SOILS FROM THE SEVEN NUCLEAR POWER STATIONS

C.1 PATHFINDER C.1

C.2 HUMBOLDT BAY C.15

C.3 DRESDEN UNIT ONE C.32

C.4 INDIAN POINT UNIT ONE C.42

C.5 MONTICELLO C.60

C.6 TURKEY POINT C.73

C.7 RANCHO SECO C.87

C.1 RESIDUAL RADIONUCLIDE CONCENTRATIONS AND INVENTORIES AT PATHFINDER

TABLE C.1.1. Residual Radionuclide Concentrations on Piping and Hardware Surfaces at Pathfinder, September 1980

pCi/cm^2 - (pCi/gm) in parentheses

Sample	Identification	^{22}Na	^{54}Mn	^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
MSL-A	Main steam line 16" carbon steel	<0.36 (<0.026)	<0.39 (<0.027)	523 (37.7)	4.28 (0.308)	1369 (98.6)	1080 (77.5)	<12 (<0.86)	<0.29 (<0.021)	<2.6 (<0.19)	3.43 (0.247)	<1.5 (<0.104)	<0.61 (<0.044)
MSB-B	Main steam bypass line	*	*	994 (205)	8.13 (1.68)	5666 1170	2061 424	*	<0.18 (<0.037)	*	*	*	*
WP-74-301A	Reactor feedwater line 8" dia. carbon steel	*	*	615 (96.4)	14.1 (2.20)	19350 (3016)	6346 (989)	*	<0.51 (<0.079)	*	*	*	*
RFW-A	Reactor feedwater line 8" dia. carbon steel	*	*	1465 (150)	15.0 (1.54)	21219 (2177)	4618 (474)	*	<0.170 (<0.017)	*	*	*	*
WP-42 11	Reactor feedwater line 8" diameter carbon steel	<0.092 (<0.014)	<0.0820 (<0.013)	1465 (228)	15.0 (2.34)	21219 (3308)	4618 (720)	<0.250 (<0.039)	<0.067 (<0.0103)	<0.59 (<0.093	<0.096 (<0.015)	<0.40 (<0.062)	<0.16 (<0.026)
WP-76-300-A3	Reactor water purification line	<1.5 (<0.34)	<1.5 (<0.35)	923 (210)	2.32 (0.531)	10117 (2309)	722 (165)	<78 (<18)	<1.2 (<0.27)	10 (<2.4)	<0.83 (<0.19)	<5.8 (<1.3)	<2.5 (<0.57)
IFDP	Inlet line to reactor water demin. purification system	<0.42 (<0.090)	<0.70 (<0.15)	*	*	1806 (387)	*	<5.9 (<1.3)	<0.372 (<0.080)	<3.3 (<0.71)	<0.26 (<0.056)	<1.8 (<0.39)	<0.79 (<0.17)
RSA	Reactor sump pump line 1 7/8" diameter stainless steel	<0.020 (<0.0090)	<0.021 (<0.0096)	*	*	33.2 15.1	*	<0.31 (<0.14)	<0.017 (<0.0077)	<0.16 (<0.071)	5.18 (2.34)	<0.079 (<0.036)	<0.042 (<0.019)
RSE	Reactor sump pump line carbon steel elbow	<0.44 (<0.043)	<0.47 (<0.045)	235 (23)	1.32 (0.13)	1669 (161)	445 (42.8)	<16 (<1.5)	<0.017 (<0.0016)	<3.2 (<0.31)	69.8 (6.71)	<1.8 (<0.17)	<0.81 (<0.077)
RSE-Pipe	Reactor sump pump line 1 1/2" dia. SS pipe from RSE	<0.076 (<0.025)	<0.14 (<0.48)	77.9 (26.0)	0.223 (0.0743)	120 (40.0)	61.3 (20.4)	<1.0 (<0.34)	<0.023 (<0.0077)	<0.50 (<0.16)	108 (40.0)	<0.21 (<0.070)	<0.19 (<0.062)
RLL-A	Reactor liquid level column 2 3/8" dia. stainless steel	<0.75 (<0.15)	<0.83 (<0.17)	14996 (2999)	11.8 (2.36)	4020 (804)	4163 (832)	<31 (<6.2)	<0.014 (<0.0027)	<5.6 (<1.1)	<0.45 (<0.091)	<3.2 (<0.63)	<1.4 (<0.27)
SPC-A	Shield pool cleanup line 4" dia. carbon steel	<0.203 (<0.082)	<0.23 (<0.093)	2803 (1135)	1.47 (0.595)	782 (317)	480 (195)	<5.1 (<2.09)	<0.011 (<0.0045)	<1.6 (<0.64)	<0.13 (<0.051)	<1.3 (<0.53)	<0.38 (<0.16)
PWC-B	Pool water cleanup line 2" dia. stainless steel (storage shield pools)	<0.68 (<0.27)	<0.73 (<0.29)	2544 (1006)	5.36 (2.13)	23601 (9335)	1646 (651)	<21 (<8.2)	<0.031 (<0.012)	<4.95 (<1.95)	<0.38 (<0.15)	<2.8 (<1.1)	<1.2 (<0.46)

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* Not measured.

TABLE C.1.1. (contd)

Radionuclide Concentrations - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{226}Ra	^{238}Pu	^{239}Pu	^{241}Am
MSL-A	Main steam line 16" carbon steel	<0.23 (<0.017)	<0.18 (<0.013)	<0.43 (<0.031)	<0.84 (<0.060)	2.52 0.181	<4.8 (<0.34)	<0.76 (<0.055)	<0.54 (<0.039)	<4.3 (<0.31)	0.00143 (0.000104)	0.00155 (0.00011)	*
MSB-B	Main steam bypass line	*	*	0.0822 (0.0169)	*	*	*	*	*	*	0.00272 (0.000562)	0.00298 (0.[illegible])	*
WP-74-301A	Reactor feedwater line 8" dia. carbon steel	*	*	0.0905 0.0141	*	*	*	*	*	*	0.00119 (0.000185)	0.00215 (0.000338)	*
RFW-A	Reactor feedwater line 8" dia. carbon steel	*	*	<0.039 (<0.0040)	*	*	*	*	*	*	0.000964 (0.0000988)	0.00778 (0.000183)	*
WP-42 11	Reactor feedwater line 8" diameter carbon steel	<0.057 (<0.090)	<0.045 (<0.0072)	<0.074 (<0.0012)	<0.42 (<0.066)	<0.28 (<0.043)	<0.97 (<0.15)	<0.15 (<0.024)	<9.6 (<1.5)	<0.91 (<0.14)	0.000964 (0.000149)	0.00176 (0.000279)	*
WP-76-300-A3	Reactor water purification line	<0.87 (<0.20)	<0.72 (<0.16)	3.48 (0.793)	<3.9 (<0.88)	<4.2 (<0.97)	<19 (<4.3)	<1.8 (<0.42)	<2.2 (<0.50)	<17 (<3.9)	0.000279 (0.000063)	0.000326 (0.000072)	0.00775 (0.00177)
IFDP	Inlet line to reactor water demin. purification system	<0.27 (<0.059)	<0.23 (<0.049)	0.914 (0.196)	<3.4 (<0.73)	<1.1 (<0.23)	<5.9 (<1.3)	<0.58 (<0.12)	<0.69 (<0.15)	<5.4 (<1.1)	*	*	*
PSA	Reactor sump pump line 1 7/8" diameter stainless steel	<0.019 (<0.0086)	<0.011 (<0.0050)	<0.019 (<0.0086)	<0.058 (<0.026)	<0.048 (<0.022)	<0.26 (<0.12)	<0.027 (<0.012)	<0.031 (<0.014)	<0.24 (<0.11)	*	*	*
RSE	Reactor sump pump line carbon steel elbow	<0.28 (<0.027)	<0.22 (<0.021)	0.820 (0.079)	<1.6 (<0.16)	4.59 (0.44)	<5.8 (<0.56)	<0.41 (<0.049)	<0.67 (<0.064)	<5.3 (<0.51)	0.000392 (0.000036)	0.00248 (0.000239)	0.00117 (0.000113)
PSE-pipe	Reactor sump pump line 1 1/2" dia. SS pipe from RSE	<0.062 (<0.021)	<0.036 (<0.012)	<0.056 (<0.018)	<0.56 (<0.19)	0.85 (0.28)	<0.68 (<0.22)	<0.12 (<0.041)	<0.093 (<0.031)	<0.61 (<0.204)	0.00108 (0.000365)	0.00676 (0.00221)	*
RLL-A	Reactor liquid level column 2 3/8" dia. stainless steel	<0.47 (<0.094)	<0.39 (<0.078)	2.02 (0.405)	<3.7 (<0.73)	<2.02 (<0.404)	<10.2 (<2.03)	<0.99 (<0.20)	<1.2 (<0.24)	<9.2 (<1.9)	0.000450 (0.000090)	0.00117 (0.000225)	*
SPC-A	Shield pool cleanup line 4" dia. carbon steel	<0.13 (<0.053)	<0.11 (<0.044)	1.35 (0.545)	<0.79 (<0.32)	<0.52 (<0.21)	<2.8 (<1.1)	<0.27 (<0.11)	<0.33 (<0.13)	<2.5 (<1.0)	0.00901 (0.00369	0.0563 (0.0230	0.0649 (0.0261)
PWC-B	Pool water cleanup line 2" dia. stainless steel (storage & shield pool)	<0.41 (<0.16)	<0.34 (<0.13)	40.2 (15.9)	<1.6 (<0.62)	<1.7 (<0.66)	<9.1 (<3.6)	<0.97 (<0.39)	<1.03 (<0.41)	<8.3 (<3.3)	0.0315 (0.0126)	0.212 (0.0838)	0.104 (0.0410)

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* Not measured.

TABLE C.1.1. (contd)

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{22}Na	^{54}Mn	^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
PDO-A	Outlet from pool demin. 2" dia. SS pipe	<0.22 (<0.085)	<0.11 (<0.041)	*	*	535 (204)	*	<4.8 (<1.8)	<0.081 (<0.031)	<0.72 (<0.27)	<0.055 (<0.,021)	<0.40 (<0.15)	<0.35 (<0.14)
PDI-A	Inlet to pool demin. 2" dia. SS pipe	<0.25 (<0.104)	<0.48 (<0.196)	*	*	1386 (565)	*	<12 (<4.95)	<0.36 (<0.15)	<3.2 (<1.3)	<0.25 (<0.101)	<1.8 (<0.74)	<0.79 (<0.33)
SPCB-A	Shield pool coolant bypass 4" dia. carbon steel pipe	<0.32 (<0.069)	<0.35 (<0.075)	*	*	873 (191)	*	<8.7 (<1.9)	<0.26 (<0.057)	<2.33 (<0.509)	<0.18 (<0.0[illegible])	<1.3 (<0.29)	<0.55 (<0.12)
SPD-A	Shield pool drain pipe 4" dia. SS pipe	<3.7 (<1.6)	<3.96 (<1.7)	330 (142)	0.572 (0.2406)	4599 (1931)	173 (72.5)	<34 (<14)	<0.59 (<0.025)	<45 (<19)	<2.3 (<0.99)	<15 (<6.4)	<6.3 (<2.6)
TR-B	Tool rack from shield pool-strip of SS plate	<0.0405 (<0.0211)	<0.046 (<0.025)	*	*	123 (64)	*	<1.19 (<0.63)	<0.0546 (<0.021)	<0.33 (<0.17)	0.14 (0.075)	<0.18 (<0.097)	<0.081 (<0.043)
TR-B-NW	Nuts and washers from TR-B-stainless steel	<2.1 (<0.82)	<2.3 (<0.901)	*	*	883 (332)	*	<7.9 (<3.1)	<3.7 (<1.4)	16 (<6.4)	<1.3 (<0.51)	<9.1 (<3.5)	<8.9 (<3.3)
FHR-A	Fuel shoot support strut from shield pool - 2 1/2" dia.	<0.86 (<0.21)	<0.99 (<0.24)	55.0 (13.4)	<0.100 (<0.024)	788 (192)	35.3 (8.603)	<4.8 (<1.2)	<0.019 (<0.0050)	<6.5 (<1.6)	<0.63 (<0.15)	<5.7 (<1.4)	<1.5 (<0.37)
SHFBS-A	Superheater fuel storage rack - 1" dia. SS tubes	<0.17 (0.36)	<0.19 (<0.41)	*	*	101 (219)	*	<0.59 (<1.27)	<0.14 (<0.31)	<1.3 (<2.9)	<0.103 (<0.22)	<0.72 (<1.5)	<0.32 (<0.69)
SHFBS-C	Same as SHFBS-A, except from another bundle	<0.53 (<1.1)	<0.91 (<1.97)	*	*	178 (386)	*	<2.7 (<5.9)	<0.45 (<0.98)	4.1 (<8.9)	<0.32 (<0.69)	<2.5 (<5.3)	<1.2 (<2.5)
FTC	Fuel transfer chute in fuel storage basin - SS plate	<0.16 (<0.089)	<0.302 (<0.17)	67.6 (37.4)	<0.19 (<0.105)	963 (534)	22.9 (12.7)	<5.7 (<3.1)	<0.071 (<0.039)	<1.20 (<0.67)	<0.17 (<0.094)	<0.68 (<0.37)	<0.43 (<0.24)
FTTR	Fuel transfer chute roller wheel - SS roller	<0.59 (<0.14)	<0.67 (<0.15)	*	*	547 (127)	*	<2.1 (<0.500)	<0.504 (<0.12)	<4.5 (<1.04)	0.91 (0.21)	<2.5 (<0.57)	<1.04 (<0.24)
FRSB-A	Fuel storage rack - 2" dia. SS cylinder - east end	<3.1 (<1.5)	<2.4 (<1.2)	294 (145)	0.356 (0.176)	4274 (2113)	117 (57.7)	<32 (<16)	<0.018 (<0.0090)	<16 (<7.97)	<1.3 (<0.65)	<9.01 (<4.5)	<4.04 (<1.99)
FRSB-C	Same as FRSB-A, except from west end	<3.9 (<2.02)	<3.98 (<2.04)	537 (275)	0.599 (0.308)	8252 (4234)	188 (96.5)	<48 (<25)	<0.025 (<0.013)	<27 (<14)	<2.2 (<1.1)	<15 (<7.8)	<6.6 (<3.4)

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* Not measured.

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TABLE C.1.1. (contd)

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{226}Ra	^{238}Pu	^{239}Pu	^{241}Am
PDO-A	Outlet from pool demin. 2" dia. SS pipe	<0.059 (<0.023)	<0.049 (<0.019)	0.245 (<0.0932)	<0.402 (<0.15)	<0.23 (<0.086)	<1.1 (<0.51)	<0.105 (<0.0401)	<0.15 (<0.058)	<1.2 (<0.058)	*	*	*
PDI-A	Inlet to pool demin. 2" dia. SS pipe	<0.26 (<0.108)	<0.23 (<0.094)	1.27 (0.523)	<2.3 (<0.94)	<1.06 (<0.44)	<5.9 (<2.4)	<0.47 (<0.19)	<0.77 (<0.32)	<5.4 (<2.2)	*	*	*
SPCB-A	Shield pool coolant bypass 4" dia. carbon steel pipe	<0.19 (<0.042)	<0.16 (<0.035)	<0.29 (<0.064)	<0.75 (<0.16)	<0.75 (<0.16)	<4.3 (<0.94)	<0.38 (<0.083)	<0.49 (<0.11)	<3.9 (<0.85)	*	*	*
SPD-A	Shield pool drain pipe 4" dia. SS pipe	<2.2 (<0.92)	<1.8 (<0.76)	<8.6 (<3.6)	<8.4 (<3.5)	<9.1 (<3.8)	<55 (<23)	<3.9 (<1.6)	<5.6 (<2.4)	<45 (<19)	<0.0000900 (<0.0000380)	<0.0000631 (<0.0000270)	
TR-B	Tool rack from shield pool- strip of SS plate	<0.028 (<0.015)	<0.024 (<0.013)	<0.0405 (<0.021)	<0.13 (<0.067)	<0.24 (<0.13)	<1.2 (<0.63)	<0.059 (<0.031)	<0.068 (<0.036)	<0.53 (<0.28)	*	*	*
TR-B-NW	Nuts and washers from TR-B- stainless steel	<2.6 (<0.995)	<1.3 (<0.49)	<2.0 (<0.78)	<6.4 (<2.4)	<4.8 (<1.9)	<31 (<12)	<3.0 (<1.1)	<3.5 <1.3	<27 (<10.4)	*	*	*
FHR-A	Fuel shoot support strut from shield pool - 2 1/2" dia	<0.55 (<0.13)	<0.45 (<0.11)	<0.83 (<0.201)	<2.1 (<0.504)	<2.1 (<0.51)	<12 (<2.9)	<1.6 (<0.39)	<1.4 (<0.34)	<11 (<2.7)	0.000315 (0.000801)	0.000766 (0.000180)	0.00302 (0.006721)
SHFBS-A	Superheater fuel storage rack - 1" dia. SS tubes	<0.11 (<0.24)	<0.091 (<0.20)	1.23 (2.65)	<0.46 (<1.04)	<0.43 (<0.92)	<2.3 (<5.00)	<0.24 (<0.51)	<0.38 (<0.82)	<2.1 (<4.5)	*	*	*
SHFBS-C	Same as SHFBS-A, except from another bundle	<0.36 (<0.78)	<0.41 (<0.88)	<0.51 (<1.11)	<1.3 (<2.8)	<1.5 (<3.2)	<7.7 (<17)	<0.62 (<1.3)	<0.89 (<1.9)	<6.7 (<14.5)	0.000541 (0.00113)	<0.000045 (<0.000090)	0.00838 (0.01802)
FTC	Fuel transfer chute in fuel storage basin - SS plate	<0.23 (<0.13)	<0.085 (<0.047)	1.40 (0.775)	<0.39 (<0.21)	<0.38 (<0.21)	<2.2 (<1.2)	<0.203 (<0.11)	<0.41 (<0.23)	<1.98 (<1.10)	0.00131 (0.000721)	0.0000676 (0.0000380)	*
FTTR	Fuel transfer chute roller wheel - SS roller	<0.38 (<0.088)	<0.304 (<0.0703)	<0.58 (<0.13)	<1.7 (<0.399)	<1.4 (<0.33)	<8.2 (<1.9)	<0.68 (<0.16)	<0.93 (<0.21)	<7.4 (<1.7)	0.00189 (0.000450)	0.0000901 (0.0000225)	*
FRSB-A	Fuel storage rack - 2" dia. SS cylinder - east end	<1.3 (<0.66)	<1.1 (<0.56)	7.7 (3.8)	<5.9 (<2.98)	<5.99 (<2.95)	<29 (<14)	<2.8 (<1.4)	<3.4 (<1.7)	<25.6 (<12.7)	0.00946 (0.00444)	0.00149 (0.000766)	*
FRSB-C	Same as FRSB-A, except from west end	<2.9 (<1.5)	<1.9 (<0.97)	25.3 (13.00)	<9.99 (<5.1)	<12 (<5.98)	<49 (<25)	<4.7 (<2.4)	<5.7 (<2.9)	<44 (<23)	0.0296 (0.0152)	0.00401 (0.00206)	*

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* Not measured.

TABLE C.1.1. (contd)

Radionuclide Concentration - $\mu Ci/cm^2$ ($\mu Ci/gm$) in parentheses

Sample	Identification	^{22}Na	^{54}Mn	^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
FSBTT	Fuel transfer tube from fuel storage basin - 2" dia.	<0.71 (<0.16)	<0.78 (<0.18)	*	*	769 (178)	*	<3.7 (<0.87)	<0.59 (<0.14)	<5.3 (<1.2)	<0.42 (<0.097)	<4.2 (<0.97)	<1.3 (<0.29)
HSHT-Pipe	High solids holdup tank discharge line - 1 1/2" dia. SS pipe	<22 <(8.9)	<16 (<6.6)	886 (362)	0.34 (0.14)	13393 (5474)	119 (49)	<124 (<51)	<0.0300 (<0.012)	<108 (<44)	<8.9 (<3.6)	<60 (<25)	<26 (<11)
HSHT-Elbow	Carbon steel elbow from discharge line from HSHT	<1.3 (<0.56)	<1.5 (<0.64)	887 (389)	1.23 (0.5405)	9351 (4104)	402 (176)	<7.3 (<3.2)	<0.031 (<0.014)	<8.8 (<3.9)	<0.71 (<0.31)	<4.9 (<2.1)	<2.8 (<1.2)
CWTD-A	Concentrated waste tank discharge line - 1 1/2" dia. SS pipe	<0.029 (<0.013)	<0.041 (<0.018)	*	*	57.5 (24.[illegible])	*	<0.401 (<0.17)	<0.025 (<0.011)	<0.23 (<0.098)	<0.018 (<0.0080)	<0.12 (<0.054)	<0.056 (<0.024)
SRTD	Spent resin tank discharge line - 2" dia. SS pipe	<0.403 (<0.12)	<0.63 (<0.19)	*	*	930 (275)	*	<8.7 (<2.6)	<0.33 (<0.098)	<2.99 (<0.88)	<0.46 (<0.14)	<2.5 (<0.75)	<0.703 (<0.21)
SCDL	Steam condensate drain line 1" dia. carbon steel line	<0.036 (<0.013)	<0.041 (<0.015)	*	*	67.6 (24.8)	*	<0.46 (<0.17)	<0.032 (<0.012)	<0.28 (<0.103)	0.0864 (0.0317)	<0.16 (<0.058)	<0.069 (<0.025)
Estimated conc. for Rad-waste system SS piping		<0.5	<0.5	300	2	5000	200	<5	<0.05	<10	<0.5	<2	<0.7
Estimated conc. for Rad-waste system carbon steel piping		<0.2	<0.2	1500	2	10000	400	<5	<0.05	<10	<0.5	<2	<0.7

* Not measured.

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TABLE C.1.1. (contd)

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ra	^{238}Pu	^{239}Pu	^{241}Am
FSBTT	Fuel transfer tube from fuel storage basin - 2" dia.	<0.45 (<0.104)	<0.36 (<0.083)	42.3 (9.82)	<2.9 (<0.67)	<1.7 (<0.39)	<16 (<3.6)	<0.804 (<0.19)	<1.1 (<0.25)	8.6 (<1.99)	*	*	*
HSHT-Pipe	High solids holdup tank discharge line - 1 1/2" dia. SS pipe	<9.01 (<3.7)	<9.2 (<3.8)	72.7 (29.7)	<40 (<16)	<51 (<21)	<234 (<95)	<32 (<13)	<23 (<9.2)	<60 (<25)	2.88 (1.18)	0.438 (0.179)	*
HSHT-Elbow	Carbon steel elbow from discharge line from HSHT	<0.74 (<0.32)	<0.61 (<0.27)	40.2 (17.6)	<3.3 (<1.4)	<3.7 (<1.6)	<16 (<6.9)	<1.5 (<0.68)	<1.8 (<0.81)	<14 (<6.3)	0.153 (0.0672)	1.02 (0.449)	0.641 (0.282)
CWTD-A	Concentrated waste tank discharge line - 1 1/2" dia. SS pipe	<0.019 (<0.0083)	<1.58 (<0.68)	0.236 (0.102)	<0.088 (<0.038)	<0.069 (<0.0300)	<0.401 (<0.17)	<0.042 (<0.018)	<0.047 (<0.0203)	<0.36 (<0.16)	*	*	*
SRTD	Spent resin tank discharge line - 2" dia. SS pipe	<0.25 (<0.073)	<0.203 (<0.0601)	<0.38 (<0.11)	<0.95 (<0.28)	<0.94 (<0.28)	<5.5 (<1.6)	<0.56 (<0.16)	<0.63 (<0.19)	<5.02 (<1.5)	*	*	*
SCDL	Steam condensate drain line 1" dia. carbon steel line	<0.024 (<0.0088)	<0.0197 (<0.0072)	<0.066 (<0.024)	<0.18 (<0.066)	<0.0807 (<0.033)	<0.445 (<0.18)	<0.059 (<0.022)	<0.059 (<0.021)	<0.45 (<0.17)	*	*	*
Estimated conc. for Rad-waste system SS piping		<0.3	<0.2	10	<1	<1	<1	<1	<1	<1	0.05	0.05	*
Estimated conc. for Rad-waste system carbon steel piping		<0.3	<0.2	10	<1	<1	<1	<1	<1	<1	0.05	0.05	*

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* Not measured.

TABLE C.1.2. Carbon-14 and ^{99}Tc Concentrations in Pathfinder Piping

Sample	Concentration (pCi/cm^2)	
	^{14}C	^{99}Tc
Main Steam Line	1.6 ± 0.5	< 0.3
Reactor Feedwater Lines	< 0.3	< 0.3
Shield Pool Coolant Line	< 0.3	< 0.3
Reactor Feedwater Line	< 0.3	< 0.3

TABLE C.1.3. Strontium-90 Concentrations in Pathfinder Piping

Sample	^{90}Sr Concentration	
	pCi/cm^2	(pCi/g)
Main Steam Line	<0.9	(<0.06)
Reactor Feedwater Line	<2	(<0.2)
Reactor Water Purification Line	<0.2	(<0.04)
Pool Water Clean-up Line	45	(18)
High Solids Holdup Tank	796	(325)
Fuel Storage Rack (Storage Basin)	0.20	(0.10)

TABLE C.1.4. Radionuclide Concentrations in Top Two Centimeters of Concrete Cores Collected at Pathfinder, September 1980

		pCi/cm^2 (a)					
Sample	Surface Condition	Depth (cm)	^{60}Co	^{134}Cs	^{137}Cs	^{152}Eu	^{154}Eu
PCC-1	clear sealer, rough surface	0-1	87.8	4.37	782.0	<0.04	<0.04
		1-2	<0.07	<0.05	0.28	<0.03	<0.04
PCC-2	thick epoxy coating	0-1	353.0	<0.27	5.59	<0.23	<0.23
		1-2	<0.02	<0.01	<0.01	<0.02	<0.01
PCC-3	clear sealer, smooth	0-1	5.63	<0.01	0.054	0.032	<0.02
		1-2	<0.05	<0.05	<0.03	<0.03	<0.03
PCC-4	clear sealer, smooth	0-1	1.49	<0.02	<0.03	0.036	<0.02
		1-2	<0.05	<0.05	<0.05	<0.03	<0.02
PCC-5	thick epoxy coating	0-1	2.12	<0.01	<0.02	<0.02	<0.02
		1-2	<0.05	<0.05	<0.04	<0.02	<0.03
PCC-6	thick epoxy coating	0-1	3.72	<0.09	0.050	<0.03	<0.02
PCC-7	gray paint, smooth	0-1	466.0	<0.08	0.29	0.27	<0.08
		1-2	8.90	<0.02	<0.02	0.086	0.031
PCC-8	chipped gray paint, clear sealer	0-1	477.0	<0.08	0.46	1.33	0.23
		1-2	10.6	<0.02	<0.02	1.31	0.22
PCC-9A	chipped gray paint, clear sealer	0-1	350.0	<0.09	<0.36	<0.18	<0.09
PCC-9B	chipped gray paint, clear sealer	0-1	294.0	<0.01	0.39	0.33	<0.06
PCC-11	clear sealer, smooth	0-1	2.39	<0.01	0.072	<0.01	<0.02
PCC-12	clear sealer, smooth	0-1	2.39	<0.01	0.072	<0.01	<0.02
		1-2	<0.03	<0.03	<0.03	<0.02	<0.02
PCC-13	clear sealer, rough surface	0-1	16.2	<0.01	<0.09	<0.02	<0.02
		1-2	<0.05	<0.04	<0.04	<0.02	<0.03
PCC-14	gray paint, clear sealer	0-1	0.14	<0.02	<0.06	<0.04	<0.02
		1-2	<0.02	<0.01	<0.01	<0.01	<0.01
PCC-15	thick epoxy coating	0-1	5.09	<0.01	0.17	<0.09	<0.02
		1-2	<0.02	<0.01	<0.01	<0.01	<0.01
PCC-16	gray paint, smooth	0-1	<0.045	<0.01	0.032	0.008	<0.02
		1-2	<0.05	<0.05	<0.04	<0.02	<0.03
PCC-17	gray paint, smooth	0-1	0.98	<0.01	0.14	0.035	<0.02
		1-2	<0.06	<0.05	<0.04	<0.03	<0.03
PCC-18	gray paint, smooth	0-1	1.16	<0.01	<0.090	0.020	0.034
PCC-19	gray paint, smooth	0-1	1.28	<0.01	0.045	0.01	<0.02
		1-2	<0.02	<0.02	<0.03	<0.01	<0.01
PCC-20	gray paint, smooth	0-1	9.05	<0.02	1.91	0.47	<0.02
		1-2	<0.05	<0.05	<0.04	<0.02	<0.03
PCC-21	gray paint, smooth	0-1	0.39	<0.01	0.068	0.017	<0.02

(a) To convert to pCi/g, multiply pCi/cm^2 by 0.472

TABLE C.1.5. Depth Distribution of Radionuclides in Concrete Cores Collected at Pathfinder, September 1980

Core	Depth (cm)	pCi/cm^2 (a)				
		^{60}Co	^{134}Cs	^{137}Cs	^{152}Eu	^{154}Eu
PCC-7	0-1	466	<0.08	0.29	0.272	<0.08
	1-2	8.90	<0.02	<0.02	0.086	0.031
	2-3	8.09	<0.01	<0.02	0.065	<0.01
	3-4	11.2	<0.01	<0.02	0.030	<0.01
	4-5	6.63	<0.01	<0.02	0.034	<0.01
	5-6	0.14	<0.009	<0.01	0.040	<0.008
	6-7	0.21	<0.01	<0.01	0.041	<0.009
	7-8	0.23	<0.009	<0.01	0.026	<0.009
	8-9	0.31	<0.009	<0.01	0.039	<0.009
	9-10	0.26	<0.009	<0.02	0.027	<0.009
	10-11	0.12	<0.009	<0.01	0.035	<0.008
	11-12	0.23	<0.01	<0.02	0.037	<0.009
PCC-8	0-1	447	<0.08	0.46	1.33	0.23
	1-2	10.6	<0.02	<0.02	1.31	0.22
	2-3	5.46	<0.01	<0.02	1.43	0.25
	3-4	5.61	<0.02	<0.02	1.50	0.22
	4-5	2.99	<0.02	<0.02	1.50	0.21
	5-6	3.05	<0.02	<0.02	1.90	0.26
	6-7	2.30	<0.01	<0.01	1.76	0.26
	7-8	2.45	<0.02	<0.02	1.85	0.27
	8-9	2.03	<0.02	<0.02	1.76	0.26

(a) To convert to pCi/g, multiply pCi/cm^2 by 0.472.

TABLE C.1.6. Radionuclide Concentrations in Soil, Cooling Tower and Boiler Sludge from Pathfinder Generating Plant, July 1980

pCi/gm

Sample	Type	Depth (cm)	^{7}Be	^{40}K	^{51}Cr	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{95}Zr	^{95}Nb	^{103}Ru	^{106}Ru	^{125}Sb	^{134}Cs
S-1	Soil	0-2.5	<0.16	10.5	<0.74	0.0051	<0.013	0.521	<0.016	<0.026	<0.052	<0.034	0.10	<0.020	<0.0037
S-2	Soil	0-2.5	<0.19	7.90	<0.87	0.0069	<0.017	1.288	<0.020	0.098	<0.063	<0.042	<0.050	<0.024	<0.0045
S-2	Soil	2.5-7.6	0.33	8.38	<1.2	<0.0051	<0.014	0.021	<0.016	0.028	<0.054	<0.039	0.078	<0.022	<0.0040
S-3	Soil	0-2.5	<0.24	12.3	<1.1	0.0059	<0.016	0.024	<0.020	0.033	<0.069	<0.050	<0.054	<0.027	<0.0051
S-3	Soil	2.5-7.6	<0.26	12.4	<1.2	<0.0067	<0.018	0.022	<0.019	0.10	<0.073	<0.051	<0.057	<0.028	<0.0052
S-4	Soil	0-2.5	<0.41	15.4	<1.7	<0.0082	<0.022	0.045	<0.026	<0.044	<0.093	<0.082	<0.082	<0.042	<0.0077
S-4	Soil	2.5-7.6	<0.31	16.5	<1.4	0.012	<0.020	0.013	<0.024	0.13	<0.091	<0.065	<0.068	<0.035	<0.0066
S-5	Soil sample from effluent pitch	0-15	1.14	8.45	<2.3	<0.013	<0.033	<0.011	<0.043	<0.067	<0.15	<0.097	<0.10	<0.058	<0.011
S-6	Soil sample from water treatment eff. backwash basin	0-5	1.00	15.6	<6.6	0.063	<0.076	1.734	<0.098	0.21	<0.36	<0.20	0.36	<0.13	<0.023
S-7	Soil	0-2.5	<0.22	9.63	<1.0	<0.0065	<0.016	0.027	<0.020	0.082	<0.064	<0.045	<0.057	<0.029	<0.0053
S-7	Soil	2.5-7.6	<0.20	13.9	<0.87	<0.0057	<0.014	0.044	<0.019	0.046	<0.059	<0.042	<0.10	<0.026	<0.0050
S-8	Soil from fenced-in storage area	0-2.5	<0.16	7.95	<0.70	<0.0048	<0.012	0.019	<0.016	0.043	<0.048	<0.050	<0.040	<0.021	<0.0038
S-9	Soil	0-2.5	0.43	10.8	<1.1	<0.0062	<0.016	<0.007	<0.020	<0.033	<0.064	<0.051	0.086	<0.029	<0.0050
S-9	Soil	2.5-7.6	<0.25	15.1	<1.2	<0.0073	<0.020	<0.009	<0.024	0.074	<0.078	<0.054	<0.061	<0.032	<0.0059
S-10	Soil	0-2.5	<0.24	3.43	<1.1	<0.0070	<0.012	0.208	<0.014	<0.024	<0.049	<0.050	0.15	<0.034	<0.0049
CT-1	Sludge from cooling tower		<0.79	19.9	<3.6	<0.021	<0.055	<0.021	<0.080	<0.11	<0.26	<0.16	0.48	<0.10	<0.020
CT-2	Iron oxides from cooling tower		<0.18	1.30	<1.1	<0.0050	<0.014	0.016	<0.016	<0.027	<0.061	<0.038	<0.044	<0.024	<0.0044
CT-3	Wood slats from cooling tower		<1.5	25.62	<9.4	<0.039	<0.11	0.192	<0.12	0.58	<0.43	<0.31	<0.35	<0.18	<0.034
FBS-1	Fossil side-boiler sludge		<3.7	5.00	<15.0	0.21	<0.45	275.0	<1.5	1.50	<1.6	<0.85	<1.1	<0.51	<0.095
FBS-2	Fossil side-boiler sludge		<3.6	4.34	<19.0	<0.16	<0.44	334.0	<3.4	<0.74	<1.6	<0.85	<1.0	<0.50	<0.091
FBS-3	Fossil side-boiler sludge		<3.9	4.15	<17.0	<0.17	<0.47	256.0	<1.9	<0.81	<1.7	0.91	<1.2	<0.52	<0.096

TABLE 3.1.6. (contd)

Sample	Type	Depth (cm)	^{137}Cs	^{141}Ce	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{228}Th	^{226}Ra	^{228}Ra	^{238}Pu	$^{239-240}$Pu	^{241}Am
S-1	Soil	0-2.5	0.253	<0.071	0.035	<0.035	<0.026	<0.043	0.303	0.261	0.355	0.00041	0.00416	0.00159
S-2	Soil	0-2.5	0.224	0.14	0.044	<0.041	0.055	<0.045	0.397	0.219	0.376	0.00026	0.00290	0.00223
S-2	Soil	2.5-7.6	0.153	<0.089	<0.032	<0.040	<0.027	<0.017	0.381	0.202	0.341			
S-3	Soil	0-2.5	0.905	0.180	0.073	<0.051	<0.031	<0.077	0.486	0.495	0.494			
S-3	Soil	2.5-7.6	0.770	0.33	0.078	<0.055	<0.033	<0.075	0.811	0.494	0.622			
S-4	Soil	0-2.5	4.38	0.21	0.18	<0.082	<0.042	<0.111	0.680	0.653	0.743	0.00408	0.0698	0.00698
S-4	Soil	2.5-7.6	1.28	0.14	<0.044	<0.064	<0.040	<0.091	0.838	0.651	0.847			
S-5	Soil sample from effluent pitch	0-15	0.016	0.45	<0.069	<0.099	<0.062	<0.058	0.892	2.07	0.395			
S-6	Soil sample from water treatment eff. backwash basin	0-5	0.439	1.51	<0.20	<0.29	<0.14	<0.366	2.57	9.62	3.79	0.000027	0.00059	0.00022
S-7	Soil	0-2.5	0.455	0.12	0.079	<0.054	<0.033	<0.020	0.905	0.635	0.788			
S-7	Soil	2.5-7.6	0.720	<0.085	<0.033	<0.049	<0.030	<0.017	0.649	0.477	0.667			
S-8	Soil from fenced-in storage area	0-2.5	0.095	<0.076	<0.030	<0.038	<0.028	<0.016	0.441	0.293	0.439			
S-9	Soil	0-2.5	0.667	<0.11	<0.041	<0.054	<0.034	<0.082	0.797	0.495	0.689			
S-9	Soil	2.5-7.6	0.269	<0.11	<0.042	<0.059	<0.041	<0.070	0.730	0.494	0.721			
S-10	Soil	0-2.5	2.89	0.10	<0.10	<0.049	0.045	<0.016	0.327	0.293	0.327			
CT-1	Sludge from cooling tower		0.662	1.24	0.450	<0.18	<0.11	<0.075	1.32	4.43	2.00			
CT-2	Iron oxides from cooling tower		0.027	0.28	<0.038	<0.044	0.059	<0.016	0.256	0.743	0.309			
CT-3	Wood slats from cooling tower		0.274	1.29	<0.23	<0.35	0.421	<0.183	0.357	1.48	0.937			
FBS-1	Fossil side-boiler sludge		<0.10	0.98	<0.32	0.86	<1.0	<0.329	0.532	0.353	1.90			
FBS-2	Fossil side-boiler sludge		0.240	3.09	<0.41	1.95	<1.0	<0.14	0.626	0.191	1.00			
FBS-3	Fossil side-boiler sludge		0.545	3.2	0.68	1.66	<1.0	<0.15	0.272	0.477	1.42			

TABLE C.1.7. Radionuclide Inventories in Pathfinder Piping and Components, September 1980

System	Material Type	Nominal O.D. (in.)	Total Length (ft)	Total Weight (lb)	Total Inside Surface Area (cm^2)	Inventory of Detectible Radionuclides-Microcuries									
						^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{108m}Ag	^{137}Cs	^{152}Eu	^{238}Pu	$^{239-240}$Pu	^{241}Am
Main steam line (used MSL-A)	carbon steel	16	132	10,952	481,000	252	2.1	658	519	1.6	<0.21	1.2	0.00069	0.00075	0.00075*
	carbon steel	12	62	3318	180,000	94	0.7	246	194	0.6	<0.08	0.5	0.00026	0.00028	0.00028*
			TOTAL			346	2.8	904	713	2.2	<0.29	1.7	0.00095	0.00103	0.00103*
Steam emergency bypass line (used MSL-A)	carbon steel	16	11	910	40,100	39	0.33	227	82	0.14	0.003	0.018	0.00011	0.00012	0.00012*
	carbon steel	14	66	4,161	255,000	219	1.85	1,270	460	0.89	0.021	0.101	0.00062	0.00067	0.00067*
	carbon steel	12	19	1,021	55,400	54	0.45	373	113	0.19	0.005	0.025	0.00015	0.00017	0.00017*
	carbon steel	8	37	1,052	22,800	22	0.19	129	47	0.078	0.002	0.010	0.00006	0.00007	0.00007*
	carbon steel	4	22	231	21,000	20	0.17	119	43	0.072	0.002	0.009	0.00006	0.00006	0.00006*
			TOTAL			354	2.99	2,750	745	1.35	0.033	0.163	0.0010	0.00110	0.00110*
Reactor feedwater lines (used WP-42 all)	carbon steel	8	693	19,794	1,346,000	1,970	20	28,600	6,220	<0.100	<0.38	0.0013	0.0024	<0.0100	<0.010*
	carbon steel	8	115	2,194	170,000	250	26	3,610	790	<0.016	<0.013	<0.048	0.0002	0.0003	0.0003*
	carbon steel	4	100	1,075	97,500	140	15	2,070	450	<0.009	<0.007	<0.027	0.0001	0.0002	0.0002*
	carbon steel	3	218	1,652	163,000	240	24	3,460	750	<0.016	<0.012	<0.048	0.0002	0.0003	0.0003*
		TOTAL				2,600	85	37,740	8,210	<0.45	<0.13	<0.50	0.0018	0.0032	0.0032*
Reactor water purification lines (used WP-76-300-A3	stainless steel	6	544	1,024	79,700	74	0.18	1,691	368	<0.066	0.28	<0.33	0.000022	0.000025	0.00062
	stainless steel	4	90	967	87,800	81	0.20	1,863	405	<0.073	0.31	<0.37	0.000024	0.000028	0.00068
	stainless steel	3	575	2,488	428,000	395	0.99	9,082	1,977	<0.36	1.49	<1.8	0.000119	0.000139	0.00332
	stainless steel	2	43	159	21,900	20	0.06	465	101	<0.018	0.076	<0.09	0.000006	0.000007	0.00017
		TOTAL				570	1.38	13,101	2,851	<0.52	2.15	<2.6	0.000171	0.000199	0.0048
Reactor water purification demineralizer tanks (used WP-76-300-A3)	stainless steel				165,182	152	0.383	1,671	119	<0.14	0.57	<0.069	0.000046	0.000053	0.000053*
Reactor water purification coolers (used WP-76-300-A3)	stainless steel				165,560	153	0.384	1,675	119	<0.14	0.57	<0.069	0.000046	0.000053	0.000053*
Shield and storage pool purification lines (used SPD-A)	stainless steel	8"	4	99	6,792	2.3	0.004	31.2	1.18	<0.016	<0.058	<0.062	<6x10^{-6}	<4x10^{-7}	<2x10^{-7}*
	stainless steel	4"	4	41	3,709	1.3	0.002	17.1	0.64	<0.008	<0.032	<0.034	<3x10^{-6}	<2x10^{-7}	<1x10^{-7}*
	carbon steel	4"	161	1,737	157,612	442	0.23	123	75.7	<0.020	0.21	<0.082	0.0014	0.0089	0.010
	stainless steel	3"	192	1,455	143,184	48.4	0.082	659	24.8	<0.33	<1.23	<1.30	<0.00013	<9x10^{-6}	<4x10^{-6}*
Pool water cleanup line (PWC-B)	stainless steel	2"	36	131	18,092	46.0	0.097	427	29.8	<0.007	0.73	<0.030	0.00057	0.0038	0.0019
Inlet line to pool demin. (PDI)	stainless steel	2"	48	175	24,123	3.5*	0.0074*	32.5	2.3*	<0.006	0.031	<0.025	0.00004*	0.00029*	0.00014
Outlet from pool demin. (PDO)	stainless steel	2"	28	102	14,067	0.8*	0.0017*	7.5	0.53*	<0.0008	0.0034	<0.003	0.00001*	0.00007*	0.00004*
		TOTAL				544	0.424	1,297	135	<0.39	0.97	<1.5	0.0020	0.013	0.012
Shield pool (used FHR-A)	stainless steel	(26.5'I.D. 20' high)			2,060,300	113	<0.21	1623	72.9	<1.3	<1.7	<4.3	0.00066	0.0015	0.0062

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* Radionuclide not actually measured in these samples. Inventory estimated by comparing with similar piping which was measured.

TABLE 3.1.7. (contd)

System	Material Type	Nominal O.D. (in.)	Total Length (ft)	Total Weight (lb)	Total Inside Surface Area (cm)	Inventory of Detectible Radionuclides-Microcuries ^{55}Fe	^{59}Ni	^{60}Co	^{63}Ni	^{108m}Ag	^{137}Cs	^{152}Eu	^{238}Pu	$^{239-240}$Pu	^{241}Am
Fuel storage basin	stainless steel	Walls floors (used FTC)			2,563,700	173	<0.49	2,469	58.7	<0.44	3.59	<0.97	0.0034	0.00017	0.00017*
	stainless steel	Fuel racks (avg. of FRSB-A C)			4,784,200	1,990	2.28	29,960	730	<8.4	78.9	<43	0.093	0.013	0.013*
	stainless steel	Superheater fuel baskets (avg. of STFBS-A C)			49,877	0.47*	0.0005*	6.9	0.17*	<0.010	0.043	<0.25	0.000027	<0.000002	0.00042
		TOTAL				2,164	2.28	32,436	789	<8.9	82.5	<44	0.096	0.013	0.013*
Rad-waste system (used estimated average radionuclide concentration for rad-waste piping see Table 2)	stainless steel	6	10	92	15,300	4.6	0.031	76.5	3.1	<0.008	0.153	<0.015	0.00077	0.00077	0.00077*
	stainless steel	4	39	216	39,800	11.9	0.080	199	7.9	<0.020	0.398	<0.040	0.00199	0.00199	0.00199*
	stainless steel	3	14	60	10,900	3.3	0.022	54.5	2.2	<0.005	0.109	<0.011	0.00055	0.00055	0.00055*
	stainless steel	2	227	600	119,300	35.8	0.239	597	23.9	<0.060	1.19	<0.119	0.0060	0.0060	0.0060*
	stainless steel	1-1/2	821	1,715	335,000	100.5	0.670	1,675	67.0	<0.17	3.35	<0.335	0.0168	0.0168	0.0168*
	stainless steel	3/4	9	8	1,890	0.6	0.004	9.5	3.8	<0.001	0.019	<0.002	0.00009	0.00009	0.00009*
	carbon steel	2	19	70	9,660	14.5	0.019	96.6	3.9	<0.005	0.097	<0.009	0.00048	0.00048	0.00048*
	carbon steel	1-1/2	131	357	51,300	76.9	0.103	513	20.5	<0.026	0.513	<0.051	0.0026	0.0026	0.0026*
	carbon steel	1	23	39	5,920	8.9	0.012	59.2	2.4	<0.003	0.059	<0.006	0.00030	0.00030	0.00030*
	PVC	4	337	913	313,400	94.0	0.627	1,567	62.7	<0.16	3.13	<0.313	0.0160	0.0160	0.0160*
	PVC	3	99	182	69,600	20.9	0.139	348	13.9	<0.035	0.696	<0.070	0.0035	0.0035	0.0035*
	PVC	2	240	218	113,100	33.9	0.226	0.57	22.6	<0.057	1.13	<0.113	0.0057	0.0057	0.0057*
	PVC	1-1/2	606	435	242,900	72.9	0.486	365	48.6	<0.12	2.43	<0.243	0.0121	0.0121	0.0121*
	PVC	3/4	102	27	18,300	5.5	0.037	91.5	3.7	<0.009	0.183	<0.018	0.00092	0.00092	0.00092*
		TOTAL				484	2.70	5,652	286	<0.68	13.5	<1.3	0.068	0.068	0.068*
High solids holdup tank (used HSHT-pipe)	stainless steel				214,506	190	0.073	2,873	25.5	<1.9	15.6	<11	0.618	0.094	0.094*
Spent resin tank (used SPTD)					744,222	42	0.28	692	28*	<0.34	<0.28	<0.70	0.0069*	0.0069*	0.0069*
Low solids holdup tank (used SPTD)					744,222	42*	0.28*	692	28*	<0.34	<-.28	<0.70	0.0069*	0.0069*	0.0069*
Concentrated waste storage tank (used CWTD-A)					193,167	0.13*	0.00042*	11.1	0.085*	<0.0035	0.046	<0.013	0.000021*	0.000021*	0.000021*
Waste surge tank (used SPTD)					140,465	7.9*	0.053*	131*	5.3*	<0.06*	<0.05*	<0.13*	0.0013*	0.0013*	0.0013*
Waste demineralizer, tank (used SPTD)					56,546	3.2*	0.021*	52.6*	2.1*	<0.026*	<0.02*	<0.053*	0.00052*	0.00052*	0.00052*
Storage pool water cooler (used PWC-B)					77,181	196.3	0.41	1,821	127	<0.029	<0.12	<0.13	0.0024	0.0164	0.0080
Shield pool water cooler (used SPD-A)					77,181	26.1	0.044	355	13.4	<0.18	<0.66	<0.70	<0.00007	<0.000005	<0.000005*
Pool Demineralizers (used PDD-A)					166,012	5.3*	0.036*	88.8	3.6*	<0.009	0.041	<0.038	<0.00089*	0.00089*	0.00089*
					GRAND TOTAL	7953	99.5	105,400	14,273	3.6	120	1.9	0.81	0.23	0.22

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* Radionuclide not actually measured in these samples. Inventory estimated by comparing with similar piping which was measured.

TABLE C.1.8. Total Pathfinder Radionuclide Inventories

Radionuclide	Half-life (yr)	1980 Inventory[(a)] (millicuries)	Inventory at[(b)] 1967 Shutdown (millicuries)
$^{239-240}$Pu	100.41 x 10^4	105.00022	56,300.00022
^{60}Co	5.27	105	580
^{63}Ni	100	14.2	15.5
^{55}Fe	2.7	7.95	224
^{137}Cs	30.2	0.12	0.16
^{59}Ni	8 x 10^4	0.10	0.10
^{108m}Ag	130	0.0036	0.004
^{152}Eu	13.4	0.0019	0.0038
^{238}Pu	87.7	0.00081	0.0009
$^{239-240}$Pu	2.41 x 10^4	0.00023	0.00023
^{241}Am	432	0.00022	0.00022
^{65}Zn	0.668		56,300
	Total	127	57,100

(a) Excluding pressure vessel, bioshield, and concrete surfaces, and condenser now in use.
(b) Includes the radionuclide inventory of the condenser.

C.2 RESIDUAL RADIONUCLIDE CONCENTRATIONS AND INVENTORIES AT HUMBOLDT BAY

TABLE 3.7. Residual Radionuclide Concentrations In Humboldt Bay Reactor Component Systems

SAMPLE		^{22}Na	^{54}Mn	^{60}Co	^{65}Zn	^{95}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Eu	^{228}Ac	^{241}Am
	WATER SAMPLES (CONCENTRATION IN pCi/ml)																			
4	STEAM SUPPRESSION CHAMBER WATER	<0.09	<0.13	21 ± 1	<0.19	<0.41	<11	<1.8	<2.0	<5.5	<0.06	157 ± 3	2810 ± 10	5.9 ± 2.7	<3.0	<1.2	<1.0	<0.82	<0.95	<1.1
3	SPENT FUEL STORAGE POOL WATER	<0.9	1.5 ± 1.0	38 ± 2	<2.9	<0.91	<20	<3.0	<4.5	<9.6	<0.06	427 ± 5	4220 ± 20	<0.07	<6.9	<9.5	9 ± 4	<1.8	<7.3	9 ± 4
	HARDWARE SYSTEMS CONCENTRATION IN pCi/cm^2 (pCi/gm)																			
1	SPENT FUEL POOL WALL	<45 (<6)	760 ± 70 (105 ± 10)	22,100 ± 200 (3040 ± 30)	370 ± 210 (50 ± 30)	<46 (<8)	<720 (<100)	<100 (<20)	530 ± 280 (70 ± 40)	<310 (<50)	<2.0 (<0.3)	26,460 ± 180 (3640 ± 20)	262,200<900 (36,100 ± 100)	<2.3 (<0.3)	<230 (<30)	<660 (<90)	140 ± 140 (20 ± 20)	<98 (<12)	<580 (<80)	200 ± 140 (29 ± 19)
5	STEAM SUPPRESSION CHAMBER WALL	<2 (<0.3)	6.3 ± 0.3 (0.87 ± 0.04)	536 ± 3 (74 ± 1)	0.9 ± 0.5 (0.12 ± 0.07)	0.79 ± 0.48 (0.11 ± 0.07)	<4.6 (<0.6)	<0.31 (<0.04)	7.5 ± 3.5 (1.0 ± 0.5)	0.52 ± 0.28 (0.07 ± 0.04)	<0.008 (<0.001)	3.2 ± 0.5 (0.44 ± 0.07)	57.1 ± 0.8 (7.9 ± 0.1)	0.59 ± 0.39 (0.08 ± 0.05)	<1.2 (<0.2)	<1.2 (<0.2)	<0.15 (<0.02)	<0.9 (<0.12)	<1.1 (<0.2)	0.30 ± 0.17 (0.04 ± 0.02)
6	STEAM RELIEF VALVE	4.0 ± 2.9	2.1 ± 1.0	1440 ± 10	<2.8	<2.9	<28	<1.8	<21	<1.2	<0.04	6.9 ± 2.6	30 ± 3	<2.8	<6.4	<6.9	<0.73	<5.5	<6.1	<0.86
7	MAIN STEAM CONDENSER	<120 (<20)	8200 ± 100 (1130 ± 20)	142,000 ± 700 (19,500 ± 100)	<120 (<20)	<120 (<20)	<1100 (<150)	<68 (<10)	<830 (<120)	350 ± 60 (48 ± 8)	<1.4 (<0.2)	<100 (<14)	280 ± 40 (39 ± 6)	<1.9 (<0.3)	<300 (<40)	<320 (<40)	<31 (<4)	<210 (<30)	<290 (<40)	<36 (<5)
8	MAIN STEAM CONDENSER	<6.5 (<1)	57 ± 5 (7.8 ± 0.7)	1630 ± 30 (224 ± 4)	<5.0 (<0.7)	<5.4 (<0.7)	<54 (<7)	<4.0 (<0.6)	<36 (<5)	21 ± 5 (2.9 ± 0.7)	<0.11 (<0.02)	<5.4 (<0.7)	710 ± 20 (98 ± 3)	<0.11 (<0.02)	<14 (<2)	<11 (<2)	<2.2 (<0.3)	<10 (<2)	15 ± 12 (2.1 ± 1.7)	3.1 ± 2.5 (0.43 ± 0.34)
9	EMERGENCY CONDENSER	9 ± 6 (1.2 ± 0.8)	7 ± 2 (1.0 ± 0.3)	4820 ± 40 (680 ± 10)	<8.0 (<1)	6 ± 5 (0.8 ± 0.7)	<50 (<7)	<3.2 (<0.4)	45 ± 38 (6.2 ± 5.2)	16 ± 3 (2.2 ± 0.4)	<0.06 (<0.01)	8 ± 5 (1.1 ± 0.7)	15 ± 2 (2.1 ± 0.3)	<3.4 (<0.5)	<14 (<2)	<14 (<2)	3.2 ± 1.4 (0.44 ± 0.19)	22 ± 10 (3.0 ± 1.4)	<12 (<2)	7.7 ± 2.0 (1.1 ± 0.3)
10	GLAND SEAL EXHAUSTER	61 ± 33 (8 ± 5)	860 ± 30 (118 ± 4)	34,600 ± 200 (4760 ± 30)	<33 (<5)	<33 (<5)	<300 (<40)	<19 (<3)	<230 (<30)	64 ± 16 (9 ± 2)	<0.40 (<0.06)	<28 (<4)	110 ± 10 (15 ± 1)	<0.54 (<0.07)	<79 (<11)	<87 (<12)	18 ± 7 (2 ± 1)	<58 (<8)	<65 (<9)	<19 (<3)
11	CONDENSATE DEMINERALIZER	28 ± 16 (4 ± 2)	280 ± 10 (39 ± 1)	14,100 ± 100 (1940 ± 10)	<14 (<2)	<15 (<2)	<140 (<20)	<9.4 (<1)	120 ± 100 (17 ± 10)	260 ± 10 (36 ± 1)	<0.18 (<0.02)	<13 (<2)	900 ± 20 (124 ± 3)	<0.26 (<0.03)	<40 (<6)	<43 (<6)	13 ± 5 (1.8 ± 0.7)	<36 (<5)	<32 (<5)	21 ± 5 (2.9 ± 0.7)
12	OFF GAS HEPA FILTERS	<3.5 (<0.5)	<2.5 (<0.3)	116 ± 7 (16 ± 1)	<4.3 (<0.6)	<4.3 (<0.6)	<150 (<20)	<29 (<4)	<16 (<2)	<30 (<4)	0.7 ± 0.5 (0.1 ± 0.07)	<15 (<2)	33,400 ± 100 (4590 ± 10)	<0.61 (<0.08)	<5.6 (<0.8)	<18 (<3)	<17 (<2)	<7.6 (<1)	<17 (<2)	<18 (<3)
15	AIR EJECTOR	<8.7 (<30)	55 ± 6 (190 ± 20)	3080 ± 50 (10,900 ± 200)	<6.4 (<20)	<7.4 (<30)	<74 (<300)	<5.1 (<20)	64 ± 55 (230 ± 190)	22 ± 6 (80 ± 20)	<0.10 (<0.4)	13 ± 7 (46 ± 25)	77 ± 6 (270 ± 20)	<0.14 (<0.5)	<23 (<80)	<18 (<60)	5.5 ± 2.3 (20 ± 8)	23 ± 15 (80 ± 50)	29 ± 18 (100 ± 60)	25 ± 4 (90 ± 10)
16	HYDRAULIC CONTROL ROD DRIVE FILTER	430 ± 400 (110 ± 100)	2000 ± 200 (630 ± 50)	289,000 ± 2000 (85,100 ± 500)	<330 (<80)	<380 (<90)	<3300 (<900)	<210 (<50)	<2800 (<700)	670 ± 210 (170 ± 50)	<4.7 (<1)	<300 (<80)	790 ± 110 (200 ± 30)	<5.8 (<2)	<970 (<250)	<780 (<200)	<97 (<26)	<680 (<180)	<760 (<190)	630 ± 130 (160 ± 30)
	RESINS (CONCENTRATIONS IN pCi/cc OF RESIN)																			
20	SUMP DRAIN SLUDGE (RESIN BEADS)	<1200	32,000 ± 1000	1.45e^6 ± 0.1e^6	<1200	<1100	<11,000	<700	<8000	1800 ± 600	<14	10,200 ± 1100	145,000 ± 2000	<20	<3000	<33,000	<840	<2100	<2000	<500
21	SUMP DRAIN SLUDGE (RESIN BEDS)	<90	140 ± 60	82,600 ± 500	<80	<80	<800	<50	<600	100 ± 40	<1.0	480 ± 80	5200 ± 100	<1.5	<200	<220	29 ± 22	<160	<160	<38 ± 26
	MISCELLANEOUS SAMPLES (pCi/SAMPLE)																			
13	OFF GAS HEPA FILTER SLUDGE	760 ± 570	2300 ± 500	79,300 ± 2700	<840	<600	<16,000	<2900	<3400	<3100	<56	<1600	6.55e^6 ± 0.02e^6	<64	<1400	<2200	<1800	<990	4000 ± 1900	<1900
17	HYDRAULIC CONTROL ROD DRIVE FILTER	<44,000	332,000 ± 19,000	69.2e^6 ± 0.3e^{6}6	63,000 ± 45,000	<41,000	<380,000	<23,000	<280,000	<18,000	<470	<34,000	13,400 ± 9900	<680	<111,000	<180,000	<11,000	<75,000	207,000 ± 99,000	<13,000

*^{241}Am DETERMINED BY γ-SPECTROMETRY · MORE ACCURATE DETERMINATION MADE BY ALPHA SPECTROMETRY

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TABLE C.2.2. Residual Radionuclide Concentrations in Humboldt Bay Reactor System

pCi/cm^2 (pCi/g)

SAMPLE		^{22}Na	^{54}Mn	^{60}Co	^{65}Zn	^{106}Ru	^{110m}Ag	^{125}Sb	^{134}Cs	^{137}Cs	^{144}Ce	^{155}Eu	^{228}Ac
2	CABLE FROM SPENT FUEL POOL	(<9)	(40 ± 10)	(3750 ± 50)	(<30)	(<280)	(<50)	(<160)	(17,140 ± 90)	(217,700 ± 300)	(<170)	(<80)	(<110)
14	AIR EJECTOR Y STRAINER	<60 (<210)	55 ± 32 (190 ± 110)	38,100 ± 300 (131,900 ± 1000)	<130 (<450)	<200 (<700)	<110 (<400)	100 ± 50 (350 ± 180)	<30 (<100)	100 ± 30 (350 ± 100)	<70 (<240)	100 ± 30 (350 ± 100)	<300 (<1000)
18	SPARGER CLAMP BRACKET	(<9)	(50 ± 20)	(41,380 ± 70)	--	<100)	(<60)	(<30)	(<20)	(<20)	(60 ± 40)	(34 ± 16)	(<150)
19	FEEDWAYER LINE	<20 (<1)	<30 (<2)	23,300 ± 140 (1280 ± 10)	--	<160 (<10)	<100 (<5)	50 ± 40 (3 ± 2)	<30 (<2)	210 ± 20 (11 ± 1)	<50 (<3)	<30 (<2)	<230 (<12)
26	REGENERATI[illegible] HEAT EXCHA[illegible]	<3200 (<300)	76,200 ± 700 (6870 ± 60)	1.15 e^7 ± 0.01 e^7 (1.04 e^6 ± 0.01 e)		48,800 ± 37,000 (4200 ± 3300)	<2.3 e^4 (<2100)	4.6 e^4 ± 0.9 e^4 (4100 ± 800)	<6000 (<540)	20,700 ± 4700 (1900<400)	<14,000 (<1300)	<6000 (<550)	<5 e^4 (<4500)

TABLE C.2.3. Residual Radionuclide Concentrations in Humboldt Bay Reactor Component Systems

pCi/cm² (pCi/gm)

SAMPLE		^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I	$^{239\text{-}240}Pu$	^{238}Pu	^{241}Am	^{242}Cm	^{244}Cm
1	FUEL STORAGE POOL WALL	$2.20e^5 \pm 0.01e^5$ $(8.0e^4 \pm 0.0e^4)$	<0.92 (<0.3)	$3.14e^3 \pm 0.29e^3$ $(1.14e^3 \pm 0.11e^3)$	101 ± 31 (37 ± 11)	850 ± 240 (310 ± 90)	<0.5 <0.2	11.8 ± 0.2 (4.3 ± 0.1)	13.5 ± 0.3 (4.9 ± 0.1)	75 ± 1 (27.3 ± 0.4)	2.4 ± 0.3 (0.87 ± 0.11)	48.6 ± 0.8 (17.8 ± 0.3)
2	SPENT FUEL POOL CABLE	(970 ± 10)	(<2)	(1640 ± 40)	(92 ± 100)	(<5)	(<2)	(11.6 ± 0.3)	(15.5 ± 0.3)	(23.0 ± 0.7)	(<0.3)	(15.8 ± 0.6)
3	FUEL STORAGE POOL WATER(a)	19 ± 7	<0.062	16 ± 3	5.2 ± 0.5	<13	<0.03	–	–	0.010 ± 0.001	<0.0003	0.0018 ± 0.0003
5	STEAM SUPPRESSION CHAMBER	1670 ± 10 (230 ± 1)	<0.009 (<0.002)	32 ± 5 ([illegible] ± 0.7)	9.3 ± 0.4 (1.28 ± 0.05)	<22 (<3)	<0.04 (<0.001)	0.12 ± 0.01 (0.016 ± 0.001)	0.155 ± 0.008 0.021 ± 0.001	0.21 ± 0.01 (0.028 ± 0.002)	0.005 ± 0.001 (0.0006 ± 0.0001)	0.122 ± 0.002 (0.0168 ± 0.0003)
7	MAIN STEAM CONDENSER	$1.96e^6 \pm 0.01e^6$ $(2.70e^5 \pm 0.01e^5)$	<4 (<0.6)	9890 ± 650 (1360 ± 90)	230 ± 20 (31 ± 3)	<1100 (<150)	<0.9 (<0.1)	24 ± 1 (3.2 ± 0.2)	51 ± 2 (7.1 ± 0.3)	40.9 ± 0.5 (5.63 ± 0.07)	1.53 ± 0.16 (0.210 ± 0.022)	48.1 ± 0.6 (6.61 ± 0.08)
10	GLAND SEAL EXHAUSTER CONDENSATE OUTLET	$1.97e^5 \pm 0.01e^5$ $(2.70e^4 \pm 0.01e^4)$	<0.3 (<0.03)	2690 ± 30 (370 ± 4)	46 ± 1 (6.3 ± 0.2)	<130 (<20)	<0.03 (<0.004)	4.50 ± 0.06 (0.619 ± 0.01)	7.27 ± 0.08 (1.00 ± 0.01)	10.1 ± 0.1 (1.39 ± 0.02)	0.23 ± 0.03 (0.032 ± 0.004)	7.7 ± 0.1 (1.06 ± 0.01)
11	CONDENSATE DEMINERALIZER STRAINER	$8.14e^6 \pm 0.$[illegible]e^6 $(1.12e^4 \pm 0.01e^4)$	<0.7 (<0.1)	1010 ± 70 (140 ± 9)	27 ± 5 (3.8 ± 0.7)	<250 (<40)	<0.4 (<0.05)	4.98 ± 0.14 (0.68 ± 0.02)	6.05 ± 0.15 (0.83 ± 0.02)	12.1 ± 0.2 (1.66 ± 0.03)	0.27 ± 0.06 (0.037 ± 0.008)	15.8 ± 0.2 (2.17 ± 0.03)
12	OFFGAS HEPA FILTER ASSEMBLY	10,[illegible]0 ± 200 (1[illegible]80 ± 20)	<3 (<0.3)	<160 (<20)	370 ± 60 (50 ± 7)	<2000 (<300)	<3 (<0.4)	1.53 ± 0.24 (0.20 ± 0.03)	1.95 ± 0.33 (0.26 ± 0.04)	0.25 ± 0.03 (0.034 ± 0.006)	0.015 ± 0.012 (0.0020 ± 0.0016)	0.093 ± 0.020 (0.012 ± 0.003)
14	AIR EJECTOR STRAINER	$5.68e^4 \pm 0.01e^4$ $(1.97e^5 \pm 0.02e^5)$	<0.3 (<20)	$21.4e^4 \pm 0.03e^4$ $(9.38e^4 \pm 0.09e^4)$	<30 (<100)	20 ± 10 (70 ± 30)	<6 (<20)	88 ± 2 (311 ± 6)	53 ± 2 (187 ± 5)	225 ± 5 (800 ± 20)	<2 (<6)	33 ± 2 (115 ± 7)
16	CONTROL ROD DRIVE HYDRAULIC FILTER	$1.17e^6 \pm 0.02e^6$ $(2.95e^5 \pm 0.03e^5)$	<30 (<8)	56,400 ± 9000 (14,200 ± 2300)	1300 ± 100 (320 ± 30)	<2400 (<600)	<4 (<1)	270 ± 10 (68 ± 2)	190 ± 8 (48 ± 2)	553 ± 5 (139 ± 1)	8.3 ± 0.9 (2.1 ± 0.2)	98 ± 2 (24.7 ± 0.5)
18	FEEDWATER SPARGER CLAMP BRACKET	$(2.34e^4 \pm 0.01e^4)$	(<0.5)	$(1.92e^3 \pm 0.01e^3)$	(22 ± 3)	(<3)	(<2)	(13.8 ± 0.3)	(24.6 ± 0.4)	(23 ± 1)	(<0.3)	(9.6 ± 0.5)
19	STAINLESS FEED WATER LINE	4760 ± 120 (242 ± 6)	<10 (<0.5)	2030 ± 100 (103 ± 5)	105 ± 12 (5.4 ± 0.6)	<13 (<0.7)	<4 (<0.2)	72 ± 1 (3.6 ± 0.1)	59 ± 1 (3.0 ± 0.1)	117 ± 10 (6.0 ± 0.1)	<0.7 (<0.03)	13.0 ± 0.8 (0.66 ± 0.04)
21	SUMP DRAIN RESIN(b)	$5.71e^6 \pm 0.02e^6$	<20	$1.61e^5 \pm 0.03e^5$	15,300 ± 400	<4000	<2	231 ± 11	313 ± 13	709 ± 7	16 ± 2	414 ± 6
22	BIOSHIELD CONCRETE INSIDE BOTTOM	$(1.29e^4 \pm 0.010e^4)$		(46 ± 14)	(<11)	(36 ± 22)	(<0.8)	(0.012 ± 0.003)	(0.014 ± 0.005)	(<0.04)	(<0.02)	
23	BIOSHIELD CONCRETE OUTSIDE BOTTOM	$(6.00e^3 \pm 0.01e^3)$		(47 ± 7)	(36 ± 3)	(12 ± 4)	(<0.5)	(0.13 ± 0.01)	(0.106 ± 0.009)	(<0.05)	(<0.07)	
24	BIOSHIELD CONCRETE OUTSIDE TOP	$(2.01e^3 \pm 0.02e^3)$		(17 ± 5)	(<10)	(<14)	(<0.8)	(0.13 ± 0.01)	(0.113 ± 0.009)	(<0.04)	(<0.03)	
25	BIOSHIELD CONCRETE INSIDE TOP	$(3.29e^4 \pm 0.02e^4)$		(50 ± 8)	(14 ± 3)	(35 ± 11)	(<1.3)	(0.056 ± 0.005)	(0.11 ± 0.01)	(0.11 ± 0.06)	(0.09 ± 0.03)	
26	REGENERATIVE HEAT EXCHANGER	$41.5e^6 \pm 0.1e^6$ $(3.74e^6 \pm 0.01e^6)$	<60 (<5)	$9.29e^5 \pm 0.09e^5$ $(8.37e^4 \pm 0.08e^4)$	8400 ± 300 (760 ± 30)	560 ± 100 (51 ± 9)	<15 (<2)	4390 ± 50 (395 ± 5)	4480 ± 50 (403 ± 5)	8290 ± 80 (747 ± 8)	43 ± 9 (3.9 ± 0.8)	2470 ± 50 (222 ± 4)

(a) pCi/ml
(b) pCi/cc

TABLE C.2.4. Radionuclide Concentrations in Bioshield Concrete at Humboldt Bay Unit 3

pCi/g

SAMPLE		^{40}K	^{54}Mn	^{60}Co	^{65}Zn	^{95}Nb	^{106}Ru	^{125}Sb	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu
22	BOTTOM INSIDE EDGE - 66 ft	7.7 ± 0.5	0.73 ± 0.04	163 ± 1	<0.27	<0.03	<1.0	0.12 ± 0.07	4.0 ± 0.1	92.2 ± 0.5	<0.10	90 ± 1	5.5 ± 0.9	0.91 ± 0.06
23	BOTTOM OUTSIDE EDGE - 66 ft	7.3 ± 0.9	0.44 ± 0.05	354 ± 1	<0.41	0.064 ± 0.040	<1.7	0.45 ± 0.14	5.5 ± 0.2	220 ± 1	<0.23	148 ± 1	8.6 ± 0.9	1.5 ± 0.1
24	TOP OUTSIDE EDGE REFUEL FLOOR	7.7 ± 0.5	0.68 ± 0.04	131 ± 1	<0.21	0.036 ± 0.027	<1.1	0.38 ± 0.13	9.3 ± 0.2	252 ± 1	0.20 ± 0.15	30 ± 1	2.3 ± 0.5	0.33 ± 0.07
25	TOP INSIDE EDGE REFUEL FLOOR	4.5 ± 0.9	3.0 ± 0.09	564 ± 1	<0.45	<0.05	<2.3	<0.20	28 ± 1	33.4 ± 1	<0.26	140 ± 1	15 ± 1	2.2 ± 0.1

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TABLE C.2.5. Radionuclide Concentrations in Top Two Centimeters of Humboldt Bay Concrete Cores

			pCi/cm^2 [a]																
SAMPLE		DEPTH	^{155}Eu	^{144}Ce	^{126}Sn	^{125}Sb	^{108m}Ag	^{134}Cs	^{106}Ru	^{137}Cs	^{94}Nb	^{166m}Ho	^{54}Mn	^{110m}Ag	^{154}Eu	^{65}Zn	^{60}Co	^{152}Eu
1	SAND BLAST PAD	0-1	<0.2	0.58	<0.06	<0.2	<0.05	<0.07	<0.5	5.6	<0.06	<0.08	0.15	<0.16	<0.9	<0.2	2.74	<0.3
		1-2	<0.2	<0.34	<0.04	<0.12	<0.04	<0.06	<0.4	0.31	<0.04	<0.07	<0.04	<0.13	<0.33	<0.11	0.18	<0.2
2	HOT SHOP FLOOR DRAIN	0-1	<2	5.5	<0.73	3.9	<0.7	110	<4	2230	<0.3	<0.5	6.1	<1.3	5.0	<0.9	245	<1.1
		1-2	<0.05	<0.15	<0.02	0.10	<0.06	<0.03	<0.4	0.19	<0.02	<0.03	<0.02	<0.06	<0.19	<0.05	0.51	<0.09
3	HOT SHOP BACKGROUND	0-1	0.09	<0.19	<0.03	<0.09	<0.03	0.12	<0.3	2.6	<0.03	<0.05	0.14	<0.14	0.48	<0.11	7.4	<0.1
		1-2	<0.2	<0.32	<0.04	<0.13	<0.04	<0.08	<0.4	<0.06	<0.04	<0.07	<0.04	<0.12	<0.40	<0.11	<0.07	<0.2
4	RAD WASTE TANK AREA[b]	0-1	<2	7.2	<0.9	<3	<2	110	<5	3890	<0.2	<0.4	1.7	<1.0	5	<0.7	114	0.9
		1-2	<0.3	<0.66	<0.1	<1	<0.13	2.8	<0.8	68	<0.2	<0.11	0.14	<0.4	<0.9	<0.2	10.1	<0.2
5	RAD WASTE TANK AREA	0-1	<1.7	<4.4	<1.0	<3	<1	25	<5	4830	<0.1	<0.22	<0.12	<0.4	<1.2	<0.4	3.54	<0.5
		1-2	<0.13	<0.35	<0.04	<0.13	<0.04	<0.06	<0.4	0.50	<0.03	<0.07	<0.36	<0.1	<0.3	<0.09	0.105	<0.3
6	RAD WASTE BUILDING	0-1	<3.2	14	<2	<6	<2	610	<10	11,130	<0.5	<0.9	1.7	<2	6.4	<1.1	175	<2
		1-2	<0.4	<1	<0.2	<0.6	<0.2	5.2	<1	159	<0.05	<0.08	<0.05	<0.2	<0.5	<0.2	1.2	<0.3
7	RAD WASTE BUILDING	0-1	<2.0	<5.4	<3	<4	<1	310	<6	6700	<0.3	<0.5	<0.26	<0.8	3.6	<0.6	28	<0.8
		1-2	<2.0	<1.6	<0.4	<2	<0.4	12	<2	594	<0.06	<0.1	<0.063	<0.2	<0.7	<0.2	3.01	<0.4
8	REACTOR BUILDING - 66 ft	0-1	<10	<5	<2	140	<2	340	<9	4250	<2	<3	32	<4	23	<3	2200	<3
		1-2	<0.2	<0.4	<0.05	<0.2	<0.04	<0.07	<1	0.25	<0.04	<0.08	<0.04	<0.2	<0.5	<0.13	0.94	<0.2
9	REACTOR BUILDING - 66 ft	0-1	<20	<10	<2	210	<2	28	<30	1130	<2	<4	72	<20	<30	<8	15,600	<7
		1-2	<0.2	<1	<0.05	<0.2	<0.04	<0.2	<0.5	0.24	<0.04	<0.09	<0.08	<0.2	<0.5	<0.12	1.9	<0.2
10	REACTOR BUILDING - 66 ft	0-1	<20	<20	<2	410	<2	28	<30	789	<5	<5	170	23	60	<10	20,440	<9
		1-2	<0.2	<0.37	<0.05	<0.2	<0.04	<0.07	<0.4	0.20	<0.04	<0.07	<0.06	<0.2	<0.5	<0.13	1.58	<0.4
11	REACTOR BUILDING - 34 ft	0-1	<0.2	<0.48	<0.2	<0.3	<0.09	0.31	<0.8	16.5	<0.07	<0.3	<0.10	<0.4	<1.1	<0.27	18.3	<0.3
		1-2	<0.13	<0.33	<0.04	<0.2	<0.03	<0.07	<0.04	0.10	<0.04	<0.07	<0.08	<0.2	<0.4	<0.11	0.26	<0.2
12	REACTOR BUILDING - 24 ft	0-1	<0.2	<0.5	<0.1	<0.3	<0.09	1.1	<0.7	22.7	<0.1	<0.2	<0.09	<0.3	<0.9	<0.24	12.2	<0.6
		1-2	<0.2	<0.4	<0.05	<0.2	<0.04	<0.07	<0.4	0.16	<0.08	<0.06	<0.09	<0.2	<0.5	<0.13	0.90	<0.2
13	CONCRETE ROOF OVER CONC WASTE TANKS	0-1	<0.4	<1	<0.3	<0.7	<0.3	12	<2	202	<0.07	<0.5	0.14	<0.4	<1.1	<0.26	16	<0.4
		1-2	<0.2	<0.4	<0.04	<0.2	<0.04	<0.07	<0.4	0.19	<0.04	<0.2	<0.04	<0.1	<0.4	<0.12	0.17	<0.3
14	ASPHALT	WHOLE	<4	<11	<3	<8	<3	1160	<13	24,100	<0.7	<2	<0.7	3 ± 2	<7	<1.3	35	<2
15	CONDENSATE DEMIN ROOM	0-1	<0.3	<2	<0.1	<1	<0.1	0.73	<2	22	<0.1	<0.2	<1	<0.5	<1.5	<0.4	38	<0.3
		1-2	<0.2	<0.4	<0.05	<0.2	<0.05	<0.2	<0.4	0.12	<0.04	<0.07	<0.05	<0.2	<0.4	<0.1	0.26	<0.6
16	CONDENSATE DEMIN ROOM[b]	0-1	<4	<10	<2	36	<2	250	<20	4320	<2	<4	340	<20	<30	<8	14,800	<8
		1-2	<2	<5	<2	9	<1	92	<9	1510	<1	<2	45	<5	<20	<3	2700	<2
17	CONDENSATE DEMIN ROOM	0-1	<1	<3	<0.6	3	<1	26	<5	515	<1	<1	25	<3	<8	<2	1060	<2
		1-2	<0.2	<0.7	<0.05	<0.2	<0.04	<0.2	<0.5	0.95	<0.04	<0.08	<0.06	<0.2	<0.7	<0.2	2.68	<1
18	CONDENSATE PUMP ROOM	0-1	<0.2	<0.4	<0.07	<0.2	<0.07	1.2	<1	44	<0.1	<0.07	0.2	<0.2	<0.6	<0.2	12	<0.2
		1-2	<0.2	<0.4	<0.04	<0.1	<0.04	<0.1	<0.4	<0.06	<0.04	<0.07	<0.09	<0.2	<0.5	<0.2	0.29	<0.4
19	TURBINE BUILDING	0-1	<0.6	<2	<0.4	<1	<0.4	27	<2	540	<0.08	<0.2	0.3	<0.4	<2	<0.3	17.5	<0.5
		1-2	<0.2	<0.4	<0.05	<1	<0.05	<0.2	<0.4	2.1	<0.07	<0.2	<0.05	<0.4	<0.5	<0.2	1.51	<0.2
20	TURBINE BUILDING	0-1	<0.5	<0.6	<0.1	<0.3	<0.1	1.6	<0.8	37	<0.2	<0.1	<0.4	<0.3	<1	<0.2	10.0	<0.2
		1-2	<0.2	<0.4	<0.04	<0.1	<0.04	<0.07	<0.4	<0.07	<0.04	<0.07	<0.05	<0.3	<0.4	<0.2	0.24	<0.6
21	TURBINE BUILDING	WHOLE	<0.3	<0.9	<0.2	<0.5	<0.2	2.9	<1.1	73	<0.09	<0.2	0.9	0.8	<2	<0.4	28.5	<0.3
22	IN YARD NEAR STACK	0-1	<1	<2	<1	<1	<0.3	11	<2	276	<0.06	<0.1	<0.7	<0.2	<0.7	<0.2	4.2	<0.4
		1-2	<0.2	<0.4	<0.5	<0.2	<0.04	<0.07	<0.6	0.35	<0.04	<0.06	<0.1	<0.1	<0.4	<0.1	0.17	<0.6
23	CONDENSATE STORAGE TANK	0-1	<0.5	<2	<0.2	<0.7	<0.7	1.2	<3	42	<0.2	<0.4	8	<2	<4	<1	238	<0.7
		1-2	<0.5	<0.4	<0.05	<0.2	<0.04	<0.2	<0.4	0.18	<0.04	<0.07	<0.05	<0.2	<0.5	<0.2	0.32	<0.2
24	AIR EJECTOR ROOM	0-1																
		1-2	<0.2	<0.4	<0.04	<0.2	<0.04	<0.07	<0.4	<0.10	<0.03	<0.07	<0.04	<0.2	<0.4	<0.1	0.23	<0.4
25	REACTOR BUILDING REFUEL LEVEL	0-1	<1	<5	<0.5	2.3	<1	5.9	<6	126	<0.6	<1	32	<3	<9	<3	1350	<2
		1-2	<0.1	<0.3	<0.03	<0.09	<0.03	<0.05	<0.3	0.18	<0.3	<0.4	<0.07	<0.1	<0.3	<0.1	0.098	<0.2
26	INTAKE PUMPING PLATFORM	0-1	<0.2	<0.4	<0.04	<0.2	<0.05	<0.1	<0.8	1.62	<0.08	<0.06	<0.04	<0.1	<1	<0.1	0.28	<0.6
		1-2	<0.2	<0.4	<0.04	<0.1	<0.04	<0.07	<0.4	<0.06	<0.04	<0.06	<0.04	<0.1	<0.4	<0.1	0.07	<0.8
27	ACCESS CONTROL AREA	0-1	<0.5	<0.4	<0.06	<0.6	<0.2	<0.08	<0.6	8.6	<0.06	<0.1	<0.4	<0.3	<0.8	<0.2	7.8	<0.3
		1-2	<0.2	<0.6	<0.04	<0.2	<0.04	<0.07	<0.4	<0.06	<0.04	<0.07	<0.04	<0.1	<0.4	<0.1	0.14	<0.2

[a] TO CONVERT TO pCi/g MULTIPLY BY 0.406

[b] CORE TAKEN OVER CRACK IN CONCRETE FLOOR

TABLE C.2.6. ^{55}Fe, ^{63}Ni, ^{90}Sr, ^{99}Tc, and ^{129}I Concentrations in Selected Concrete Cores from the Humboldt Bay Nuclear Unit

		pCi/kg				
		^{55}Fe	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I
HBCC-4	0-1 cm	6.0×10^{6} $\pm0.1\times10^{6}$	1.9×10^{4} $\pm0.6\times10^{4}$	<950	1.9×10^{4} $\pm0.5\times10^{4}$	<60
HBCC-10	0-1 cm	93.0×10^{6} $\pm0.1\times10^{6}$	2.5×10^{6} $\pm0.1\times10^{6}$	5.1×10^{4} $\pm0.4\times10^{4}$	$<2.3\times10^{4}$	<100
HBCC-16	0-1 cm	451×10^{6} $\pm1\times10^{6}$	4.5×10^{5} $\pm0.2\times10^{5}$	8.8×10^{4} $\pm0.5\times10^{4}$	$<0.7\times10^{4}$	<80

TABLE C.2.7. Transuranic Radionuclide Concentrations in Selected Humboldt Bay Concrete Cores

		pCi/kg			
		^{238}Pu	$^{239-240}$Pu	^{241}Am	^{244}Cm
HBCC-4	0-1 cm	10 ± 5	39 ± 4	<30	<40
HBCC-10	0-1 cm	3000 ± 50	3700 ± 60	1770 ± 80	360 ± 30
HBCC-16	0-1 cm	2050 ± 60	1620 ± 50	1250 ± 80	470 ± 50

TABLE C.2.8. Removal of Radionuclides from Concrete Surfaces by Paint Stripping

Location	Core #	Surface Removed	Removal Process	% Activity Removed ^{60}Co	% Activity Removed ^{137}Cs
Rad Waste Building	6	Clear sealer	stripper	3	2
			mild abrasion	44	30
66 ft. Reactor Building	9	grey paint	stripper	39	4
		grey paint	mild abrasion	72	25
		black sealer	abrasion	92	61
Cond. Demin. Room	17	grey paint	stripper	83	66
		black sealer	mild abrasion	94	74
Cond. Storage Tank	23	clear sealer	stripper	33	24
			mild abrasion	68	45
Refuel Level Reactor Building	25	grey paint	stripper	98	98

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TABLE C.2.9. Radionuclide Concentrations in Onsite Soils Near Humboldt Bay

pCi/gm

SAMPLE	TYPE	DEPTH (cm)	^{40}K	^{51}Cr	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{95}Zr	^{95}Nb	^{103}Ru	^{106}Ru	^{125}Sb	^{134}Cs	^{137}Cs	^{141}Ce	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu
*HBSC-1-1	CORE	0-1	8.03	<0.5	0.044	<0.03	0.34	<0.07	0.19	0.35	<0.05	0.460	<0.1	<0.02	0.35	<0.06	<0.1	<0.2	<0.2	<0.08
		1-2	6.48	<0.6	0.036	<0.02	0.047	<0.06	<0.05	0.13	<0.04	<0.2	<0.1	<0.02	0.13	<0.06	<0.1	<0.2	<0.1	<0.06
		2-3	7.66	<0.4	0.024	<0.02	<0.02	<0.05	<0.05	0.11	<0.03	<0.2	<0.09	<0.02	0.16	0.10	0.1	<0.2	<0.1	<0.06
		3-4	6.74	<1.0	<0.02	<0.02	0.041	<0.04	0.070	<0.08	<0.06	<0.09	<0.05	<0.01	0.17	0.13	0.2	<0.09	<0.06	<0.03
HBSC-1-2	CORE	0-1	9.63	<0.5	<0.02	<0.03	<0.03	<0.06	0.18	0.34	<0.05	0.324	<0.1	<0.02	0.23	<0.07	<0.1	<0.2	<0.1	<0.07
		1-2	7.90	<0.4	<0.02	<0.02	<0.03	<0.06	0.12	0.23	<0.04	0.226	<0.1	<0.02	0.12	<0.07	<0.1	<0.2	<0.1	<0.07
		2-3	9.51	<0.6	<0.03	<0.03	<0.03	<0.06	<0.05	0.12	<0.04	<0.02	<0.1	<0.02	0.19	<0.06	<0.1	0.2	0.2	<0.07
		3-4	8.30	<0.7	<0.02	<0.03	0.048	<0.05	<0.05	0.078	<0.03	<0.1	<0.1	<0.02	0.15	<0.07	<0.1	<0.2	<0.1	0.13
HBSC-1-3	CORE	0-1	8.65	<0.4	<0.02	0.04	0.041	<0.06	<0.06	0.25	<0.04	<0.2	<0.1	<0.02	0.22	<0.06	<0.1	<0.2	<0.1	0.088
		1-2	8.52	<0.6	<0.02	<0.03	<0.02	<0.06	<0.06	<0.07	<0.05	0.3	<0.1	<0.02	0.20	0.09	<0.1	<0.2	<0.1	0.065
		2-3	9.15	<0.6	<0.02	<0.03	0.047	<0.06	<0.05	<0.06	<0.04	<0.2	<0.1	<0.02	0.20	<0.06	<0.1	<0.2	<0.1	0.075
		3-4	11.1	<0.5	<0.02	<0.03	0.041	<0.06	<0.05	0.11	<0.05	<0.2	<0.1	<0.02	0.37	<0.08	<0.1	<0.2	<0.1	<0.07
*HBSC-1-4	CORE	0-1	13.5	<0.9	<0.03	<0.05	0.31	<0.1	<0.1	0.62	<0.08	<0.3	<0.2	<0.04	1.10	0.29	0.40	<0.4	<0.2	0.24
		1-2	13.1	<1	0.05	<0.05	0.39	<0.09	<0.08	<0.09	<0.07	<0.3	<0.2	<0.03	0.89	<0.1	<0.2	<0.3	<0.2	<0.1
		2-3	12.0	0.9	0.04	<0.04	0.22	<0.07	<0.07	<0.08	<0.06	<0.2	<0.1	<0.03	0.64	<0.1	<0.2	0.56	<0.2	0.09
		3-4	11.6	<0.6	<0.02	<0.03	0.055	<0.06	<0.06	<0.06	<0.04	<0.2	<0.1	<0.02	0.24	0.13	<0.1	<0.2	<0.1	0.09
HBSC-1-5	CORE	0-1	14.4	<1	<0.04	<0.05	0.41	<0.1	<0.1	<0.1	<0.09	<0.4	<0.2	<0.04	1.9	<0.1	<0.2	<0.4	<0.2	0.22
		1-2	8.21	<0.7	<0.02	<0.03	0.17	<0.06	0.09	0.13	<0.06	<0.2	<0.1	<0.02	0.98	<0.09	<0.1	<0.2	<0.2	0.11
		2-3	12.6	<1	<0.03	<0.03	0.055	<0.06	0.15	0.096	<0.05	<0.2	<0.1	<0.02	0.46	<0.09	<0.1	<0.2	<0.1	<0.07
		3-4	11.2	<0.6	<0.02	<0.03	0.040	<0.06	<0.05	<0.07	<0.05	<0.2	<0.1	<0.02	0.21	<0.08	<0.1	<0.2	<0.1	0.13
*HBSC-1-6	CORE	0-1	13.3	<0.6	<0.03	<0.04	<0.03	<0.08	<0.07	0.31	<0.06	<0.2	<0.1	<0.02	0.25	<0.09	<0.1	<0.2	<0.1	0.13
		1-2	12.4	<0.7	<0.02	<0.03	<0.03	<0.06	<0.07	0.21	<0.05	<0.2	<0.1	<0.02	0.27	0.12	<0.2	<0.2	<0.1	0.078
		2-3	13.3	<3	<0.04	<0.05	<0.03	<0.07	<0.1	<0.2	<0.2	<0.2	<0.1	<0.02	0.32	0.39	<0.1	<0.2	<0.1	<0.07
		3-4	10.8	<0.6	<0.02	<0.07	<0.02	<0.06	<0.05	<0.06	<0.05	0.3	<0.09	<0.02	0.12	<0.07	<0.1	<0.2	<0.1	0.075
HBSC-1-7	CORE	0-1	9.00	<0.6	<0.02	<0.03	<0.02	<0.06	<0.05	0.074	<0.05	<0.2	<0.1	<0.02	0.064	<0.08	<0.1	<0.2	<0.1	<0.07
		1-2	8.14	<0.6	0.04	<0.03	0.030	<0.06	0.12	0.31	<0.06	<0.2	<0.1	<0.02	0.069	0.23	0.25	<0.2	<0.1	0.13
		2-3	10.1	<0.6	<0.02	<0.03	<0.02	<0.07	<0.06	0.058	<0.04	<0.2	<0.09	<0.02	0.046	<0.08	<0.1	<0.2	<0.1	<0.07
		3-4	10.4	<0.6	<0.02	<0.03	<0.02	<0.06	<0.06	<0.06	<0.08	<0.3	<0.1	<0.02	0.074	0.23	<0.1	<0.2	<0.1	<0.07

*ANALYZED FOR TRANSURANIUM NUCLIDES

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TABLE C.2.10. Transuranic Radionuclides in Selected Onsite Soils Near Humboldt Bay Generating Station

		pCi/kg					
		^{238}Pu	$^{239-240}$Pu	^{241}Am	^{242}Cm	^{244}Cm	^{238}Pu/$^{239-240}$Pu
HBSC-1-1	0 - 1 cm	0.64 ± 0.18	2.9 ± 0.2	20 ± 3	<3	8.2 ± 1.8	0.219
HBSC-1-4	0 - 1 cm	1.1 ± 0.2	11.8 ± 0.5	10 ± 2	<2	3.6 ± 0.9	0.096
HBSC-1-6	0 - 1 cm	0.55 ± 0.14	5.4 ± 0.4	4.1 ± 1.4	<0.5	<0.5	0.101

TABLE C.2.11. Radionuclide Concentrations in Selected Exclusion Area Soils at Humboldt Bay, July 1981

pCi/gm

SAMPLE	TYPE	DEPTH (cm)	^{40}K	^{51}Cr	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{95}Zr	^{95}Nb	^{103}Ru	^{106}Ru	^{125}Sb	^{134}Cs	^{137}Cs	^{141}Ce	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu
HBSG-0-1	SOIL-GRAB SAMPLE		17.4	<2	0.621	<0.2	33.0	<0.2	<0.2	<0.2	<0.2	<0.7	<0.4	1.5	24.6	<0.2	<0.3	<0.7	<0.6	<0.2
HBSG-0-3	SOIL-GRAB SAMPLE		15.7	<5	3.91	<0.4	290.0	1.72	<0.6	<0.6	<0.4	4.99	2.50	4.7	72.5	<0.5	<0.7	<1	<2	<0.4
HBSG-0-3A	SOIL-GRAB SAMPLE		17.1	<8	5.49	<0.4	377.0	0.861	0.708	<0.7	<0.6	<2	4.90	6.0	90.6	0.768	1.3	<2	3.1	<0.5
HBSC-01-1	CORE	0-2	12.7	<5	4.42	<0.3	306.0	1.14	0.656	<0.6	<0.4	<2	3.37	6.1	88.9	<0.5	<0.7	<1	2.1	<0.4
HGSC-01-2		2-6	14.3	<2	0.968	<0.2	67.8	<0.4	<0.4	<0.4	<0.3	<1	<0.5	1.2	21.4	<0.3	<0.4	<0.9	<0.9	<0.2
HGSC-01-4		6-10	16.9	<2	0.121	<0.1	16.0	<0.2	<0.2	<0.2	<0.2	<1	<0.3	0.22	4.88	<0.2	<0.3	<0.6	<0.6	<0.2
HGSC-01-6		10-14	20.3	<2	<0.07	<0.1	7.91	<0.2	0.4	<0.2	<0.1	<0.5	<0.3	<0.06	2.71	<0.2	<0.2	0.7	<0.4	<0.1
HGSC-01-8		14-18	20.3	<2	<0.08	<0.1	8.30	<0.2	<0.2	<0.2	<0.2	0.86	<0.3	0.16	2.36	<0.2	<0.3	<0.6	<0.5	<0.2
HGSC-01-10		18-22	16.5	<1	<0.05	<0.07	2.98	<0.1	<0.1	<0.1	<0.1	<0.4	<0.2	<0.04	1.24	0.3	0.4	<0.4	<0.3	0.2
HGSC-01-12		22-26	15.8	<2	<0.04	<0.08	0.109	<0.1	<0.1	<0.1	<0.1	<0.3	<0.2	<0.03	<0.03	<0.2	<0.2	<0.3	<0.2	<0.1
HGSC-01-14		26-30	10.2	<1	<0.04	<0.06	2.31	<0.1	<0.1	<0.1	<0.1	<0.3	<0.2	<0.03	0.526	<0.2	<0.2	<0.3	<0.2	<0.1
HGSC-01-16		30-34	14.4	<1	<0.04	<0.06	1.92	<0.1	<0.1	<0.1	<0.08	<0.3	<0.2	<0.03	0.304	<0.1	<0.2	<0.3	<0.2	<0.09
HGSC-01-18		34-38	14.5	<1	<0.04	<0.05	2.16	<0.09	<0.09	<0.1	<0.08	<0.3	<0.1	<0.03	0.238	<0.1	<0.1	<0.2	<0.2	0.16
HGSC-01-20		38-42	12.5	<1	<0.04	<0.1	4.71	<0.1	<0.2	<0.1	<0.1	0.5	<0.2	<0.03	0.480	<0.2	<0.2	<0.3	<0.3	<0.1
HGSC-01-22		42-46	11.5	<2	<0.07	<0.08	8.55	<0.2	<0.2	<0.2	<0.1	0.5	<0.2	<0.04	0.742	0.3	<0.2	<0.3	<0.3	<0.1
HGSC-01-24		46-50	11.2	<2	<0.08	<0.2	16.8	<0.2	<0.2	<0.3	<0.2	<0.5	<0.3	<0.05	0.970	<0.2	<0.2	<0.4	<0.5	<0.1
HBSC-02-0	CORE	0-4	11.3	<8	0.453	<0.2	26.0	<0.2	<0.3	<0.5	<0.5	<0.7	0.59	2.1	37.7	<0.7	<0.4	<0.7	<0.5	<0.2
HGSC-02-2		4-8	10.7	<9	0.614	<0.2	34.7	<0.3	0.4	<0.6	<0.5	<0.8	0.61	3.1	50.6	<2	<0.4	<0.8	1.20	0.36
HBSC-02-4		8-12	9.62	<5	0.116	<0.1	16.1	<0.2	<0.3	0.5	<0.3	<0.5	<0.3	0.76	16.3	<0.5	<0.3	<0.5	<0.4	<0.2
HBSC-02-6		12-18	5.30	<4	<0.04	<0.09	4.30	<0.1	<0.2	<0.3	<0.3	<0.3	<0.2	0.32	5.32	<0.3	<0.2	<0.3	<0.2	<0.1
HBSC-02-8		18-24	9.26	<3	<0.03	<0.06	2.03	<0.09	<0.1	0.3	<0.2	<0.2	<0.1	0.11	2.83	<0.3	<0.1	<0.2	<0.2	<0.1
HBSC-02-10		24-28	7.49	<3	0.043	<0.06	0.750	<0.1	<0.1	<0.2	<0.2	<0.3	<0.1	<0.03	1.25	<0.3	<0.2	<0.2	<0.2	<0.2
HBSC-02-12		28-32	7.08	<2	<0.02	<0.05	0.351	<0.06	<0.1	0.2	<0.1	<0.2	<0.1	<0.02	0.869	<0.2	<0.1	<0.2	<0.1	<0.07
HBSC-02-14		32-36	7.71	<3	<0.04	<0.08	3.33	<0.1	<0.2	<0.2	<0.2	<0.3	<0.1	0.03	0.320	<0.3	<0.2	<0.2	<0.2	<0.08

TABLE C.2.12. Transuranic Radionuclides in Selected Exclusion Area Soils at Humboldt Bay Generating Station

pCi/kg

		^{238}Pu	$^{239\text{-}240}Pu$	^{241}Am	^{244}Cm	$^{238}Pu/^{239\text{-}240}Pu$
HBSC-01-1	**0-2 cm**	**165 ± 1**	**193 ± 1**	**–**	**–**	**0.856**
HBSC-01-24	**46-50 cm**	**8.2 ± 0.5**	**9.5 ± 0.5**	**14 ± 1**	**5.5 ± 0.9**	**0.857**
HBSC-02-0	**0-4 cm**	**19.5 ± 0.5**	**18.2 ± 0.5**	**38 ± 4**	**11 ± 2**	**1.077**
HBSC-03A	**SURFICIAL**	**170 ± 1**	**230 ± 1**	**139 ± 3**	**28 ± 1**	**0.742**

TABLE C.2.13. Radionuclide Concentrations in Humboldt Bay Cooling Canal Sediments, July 1981

pCi/gm

SAMPLE	TYPE	DEPTH (cm)	^{40}K	^{51}Cr	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{95}Zr	^{95}Nb	^{103}Ru	^{106}Ru	^{125}Sb	^{134}Cs	^{137}Cs	^{141}Ce	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu
SED 1	CORE	0-2.5	27.9	<1	<0.05	<0.05	<0.06	<0.1	0.17	0.44	<0.08	<0.4	<0.2	<0.04	0.297	<0.1	<0.3	0.56	<0.3	0.18
		2.5-5.0	25.7	<0.8	<0.04	<0.05	0.11	<0.1	<0.1	<0.08	<0.07	0.4	<0.2	<0.04	0.287	<0.1	<0.2	<0.4	<0.3	<0.1
		5.0-7.5	29.7	<0.8	<0.04	<0.07	<0.06	<0.1	0.13	<0.1	<0.08	<0.3	<0.2	<0.04	0.482	<0.1	<0.2	<0.4	<0.3	0.27
		7.5-10.0	22.9	<0.8	<0.03	<0.04	0.06	<0.1	<0.08	<0.07	<0.06	<0.3	<0.2	<0.03	0.203	<0.09	<0.2	<0.3	<0.2	<0.1
SED 2	CORE	0-2.5	22.0	<1	0.1	<0.09	3.48	<0.2	<0.2	0.35	<0.1	<0.5	<0.3	<0.07	3.78	<0.2	0.35	<0.6	<0.4	<0.2
		2.5-5.0	16.4	<1	<0.04	<0.05	2.06	<0.1	<0.08	<0.08	<0.1	<0.3	<0.2	<0.04	1.43	<0.09	<0.2	<0.3	<0.2	<0.1
		5.0-7.5	22.8	<1	<0.05	<0.06	1.80	<0.1	<0.1	<0.09	<0.2	1.1	<0.2	0.19	5.72	<0.1	<0.2	<0.5	<0.3	<0.1
		7.5-10.0	19.7	<2	0.46	<0.1	13.9	<0.2	<0.2	<0.1	<0.2	<0.7	0.56	1.4	29.3	<0.2	<0.3	<0.7	<0.5	<0.2
SED 3	GRAB	0-5	13.2	<0.7	<0.02	<0.03	1.59	<0.08	<0.07	<0.07	<0.1	<0.3	<0.2	<0.03	7.20	<0.12	<0.26	<0.3	<0.2	<0.09
SED 4	CORE	0-2.5	17.7	<0.1	0.25	<0.08	7.55	<0.2	<0.2	0.28	<0.1	<0.5	<0.3	0.10	2.08	<0.1	<0.2	<0.4	<0.4	0.22
		2.5-5.0	18.4	<1	<0.04	<0.1	1.70	<0.1	0.28	<0.08	<0.08	<0.7	<0.2	0.11	3.35	<0.1	<0.2	<0.4	<0.3	<0.1
		5.0-7.5	12.4	<0.9	0.22	<0.06	5.82	<0.1	<0.1	<0.09	<0.09	<0.4	<0.2	0.40	6.81	<0.1	<0.2	<0.4	<0.1	<0.3
		7.5-10.0	19.6	<1	0.39	<0.1	16.4	<0.2	<0.2	<0.2	<0.1	<0.6	<0.3	0.29	4.37	<0.2	<0.3	<0.6	0.90	<0.2

TABLE C.2.14. Transuranic Radionuclide Concentrations in Humboldt Bay Generating Station Cooling Canal Sediments

pCi/kg

			^{238}Pu	$^{239\text{-}240}$Pu	^{241}Am	^{238}Pu/$^{239\text{-}240}$Pu
SEDIMENT 1	0 - 2.5 cm	INTAKE	1.1 ± 0.5	4.5 ± 0.9	100 ± 20	0.255
SEDIMENT 2	0 - 2.5 cm	EFFLUENT	2.3 ± 1.2	5.9 ± 1.4	<0.04	0.373
SEDIMENT 2	7.5 - 10 cm	EFFLUENT	6.8 ± 2.3	16 ± 3	<0.04	0.416

TABLE C.2.15. Radionuclide Inventory Estimates (millicuries) for Humboldt Bay Reactor Components - Piping Systems July 1981

	MATERIAL		NOMINAL OD. (in.)	LENGTH (m)	INSIDE SURFACE AREA (cm^2)	^{54}Mn	^{60}Co	^{65}Zn	^{125}Sb	^{134}Cs	^{137}Cs	^{155}Eu	^{55}Fe	^{63}Ni	^{90}Sr	^{99}Tc	$^{239\text{-}240}$Pu	^{238}Pu	^{241}Am	^{242}Cm	^{244}Cm
MAIN STEAM PIPING (CONDENSER)*	CARBON STEEL	SCH 80	12	43.2	3.9×10^5	3.22	56	<0.047	0.137	<0.039	0.110	<0.012	770	3.89	0.090	<0.432	0.009	0.020	0.018	6×10^{-4}	0.018
	CARBON STEEL	SCH 80	10	3.4	2.6×10^4	0.213	3.7	<0.003	0.009	<0.003	0.007	<0.001	51	0.26	0.006	<0.026	0.00062	0.001	0.001	0.4×10^{-4}	0.001
	CARBON STEEL	SCH 80	6	18.5	7.5×10^4	0.617	11	<0.009	0.026	<0.008	0.021	<0.002	147	0.74	0.017	<0.083	0.0018	0.004	0.003	1×10^{-4}	0.004
EMERGENCY CONDENSER PIPING (EMERGENCY CONDENSER/ CONDENSER)*	CARBON STEEL	SCH 80	6	38.7	1.8×10^5	0.0012	0.88	<0.002	0.003	0.002	0.003	0.0006	12	0.06	0.0014	<0.007	1.4×10^{-4}	3.1×10^{-4}	2.5×10^{-4}	9.4×10^{-6}	2.8×10^{-4}
	CARBON STEEL	SCH 80	3	10.4	2.4×10^4	0.0002	0.12	<0.0002	0.0004	0.0002	0.0004	0.00008	1.7	0.008	0.0009	<0.0009	1.9×10^{-5}	4.3×10^{-5}	3.4×10^{-5}	1.3×10^{-6}	3.9×10^{-5}
	CARBON STEEL	SCH 80	4	10.7	3.3×10^4	0.0002	0.16	<0.0003	0.0005	0.0003	0.0005	0.0001	2.2	0.011	0.0003	<0.001	2.6×10^{-5}	5.7×10^{-5}	4.6×10^{-5}	1.7×10^{-6}	5.1×10^{-5}
	STAINLESS STEEL	SCH 80	4	4.6	1.4×10^4	0.00010	0.07	<0.0001	0.0002	0.0001	0.0002	0.00005	0.96	0.005	0.0001	<0.0005	1.1×10^{-5}	2.6×10^{-5}	2.0×10^{-5}	7.5×10^{-7}	2.3×10^{-5}
CONDENSATE PIPING (Y STRAINER)	CARBON STEEL	3/8" WELL	12	10.7	1.0×10^5	0.006	3.9	<0.013	0.010	<0.003	0.010	0.010	5.68	2.19	<0.003	0.002	0.009	0.005	0.023	<0.0002	0.003
	CARBON STEEL	SCH 40	10	47.5	3.8×10^5	0.021	14.5	<0.050	0.038	<0.011	0.038	0.038	21.2	8.15	<0.011	0.008	0.034	0.020	0.086	<0.0008	0.013
REACTOR CLEAN-UP PIPING (REGENERATIVE HEAT EXCHANGER)*	STAINLESS STEEL	SCH 80	2	13.7	2.1×10^4	1.63	250	–	0.96	<0.13	0.44	<0.13	890	20	0.080	0.012	0.093	0.096	0.18	9×10^{-4}	0.053
	STAINLESS STEEL	SCH 80	1	7.0	5.4×10^3	0.41	62	–	0.24	<0.03	0.11	<0.03	220	5.0	0.045	0.003	0.024	0.024	0.046	2×10^{-4}	0.013
	CARBON STEEL	SCH 80	2	87	1.3×10^5	9.9	1500	–	5.9	<0.78	2.7	<0.78	5420	120	1.1	0.072	0.57	0.58	1.1	0.006	0.323
	CARBON STEEL	SCH 80	3	0.9	2.1×10^3	0.16	25	–	0.10	<0.01	0.04	<0.01	89	2	0.018	0.001	0.009	0.010	0.018	9×10^{-5}	0.005
SHUTDOWN PIPING (REGENERATIVE HEAT EXCHANGER)*	CARBON STEEL	SCH 40	8	18	1.1×10^5	8.5	1300	–	5.0	<0.67	2.3	<0.67	4640	104	0.94	0.063	0.485	0.501	0.939	0.005	0.28
	CARBON STEEL	SCH 40	6	35	1.7×10^5	13	1900	–	7.5	<1.0	3.5	<1.0	6960	160	1.4	0.095	0.728	0.752	1.41	0.007	0.42
	STAINLESS STEEL	SCH 120	8	2.4	1.4×10^4	1.1	160	–	0.6	<0.08	0.29	<0.08	580	13	0.117	<0.008	0.061	0.083	0.12	6×10^{-4}	0.035
FEEDWATER PIPING (FEEDWATER PIPE)*	STAINLESS STEEL	SCH 120	8	6.7	3.8×10^4	<0.0011	0.69	–	0.002	<0.0011	0.008	<0.0011	0.18	0.077	0.004	<0.001	0.003	0.002	0.006	$<3 \times 10^{-5}$	0.0005
	CARBON STEEL	SCH 120	6	2.1	9.3×10^3	<0.0003	0.22	–	0.0005	<0.0003	0.002	<0.0003	0.04	0.019	0.001	<0.0002	0.0007	0.0006	0.0012	$<7 \times 10^{-6}$	0.0001
	CARBON STEEL	SCH 100	8	7.0	4.2×10^4	<0.0012	0.98	–	0.002	<0.0012	0.009	<0.0012	0.20	0.085	0.004	<0.001	0.003	0.002	0.005	3×10^{-5}	0.0005
	CARBON STEEL	SCH 100	10	31	2.3×10^5	<0.007	5.4	–	0.12	<0.007	0.049	<0.007	1.1	0.47	0.024	<0.006	0.018	0.012	0.031	1.8×10^{-4}	0.003
RAD WASTE PIPING (REGENERATIVE HEAT EXCHANGER)*	CARBON STEEL	SCH 80	2	250	3.9×10^5	0.11	5.5	<0.008	0.10	<0.005	0.35	0.005	3200	0.39	0.011	<0.10	1.9×10^{-3}	2.4×10^{-3}	4.3×10^{-3}	1.1×10^{-4}	6.2×10^{-3}
	CARBON STEEL	SCH 40	3	160	4.0×10^5	0.11	5.6	<0.008	0.10	<0.005	0.36	0.005	3300	0.40	0.011	<0.10	2.0×10^{-3}	2.4×10^{-3}	4.8×10^{-3}	1.1×10^{-4}	6.3×10^{-3}
	CARBON STEEL	SCH 40	4	50	1.7×10^5	0.05	2.4	<0.002	0.044	<0.002	0.15	0.002	1400	0.17	0.005	<0.04	0.83×10^{-3}	1.0×10^{-3}	2.0×10^{-3}	0.45×10^{-4}	2.6×10^{-3}

*SAMPLE CONCENTRATIONS USED FOR INVENTORY ESTIMATION

TABLE C.2.16. Radionuclide Inventory Estimates (millicuries) for Humboldt Bay Reactor Components - Tanks and Heat Exchangers, July 1981

	SURFACE AREA, cm^2	^{54}Mn	^{60}Co	^{65}Zn	^{125}Sb	^{134}Cs	^{137}Cs	^{155}Eu	^{55}Fe	^{63}Ni	^{90}Sr	^{99}Tc	$^{239\text{-}240}Pu$	^{238}Pu	^{241}Am	^{242}Cm	^{244}Cm
FUEL BASIN - WALLS	3.2×10^6	2.42	70.4	1.18	<1.0	84.3	836	0.45 ± 0.62	701	10	0.32	2.70	0.038	0.043	0.24	7.6×10^{-3}	0.154
RACKS	4.4×10^6	3.35	97.3	1.63	<1.4	117	1150	0.62 ± 0.62	969	14	0.44	3.74	0.051	0.059	0.33	11×10^{-3}	0.214
MAIN CONDENSER	2.9×10^7	234	4050	<3.4	10	<3	8.0	<0.9	55,900	280	6.6	<31	0.68	1.45	1.17	0.043	1.37
EMERGENCY CONDENSER	5.0×10^5	3.5×10^{-3}	2.46	$<5 \times 10^{-3}$	8×10^{-3}	3.9×10^{-3}	7.5×10^{-3}	1.6×10^{-3}	34	0.2	0.004	<0.02	4.1×10^{-4}	8.8×10^{-4}	7.1×10^{-4}	2.6×10^{-5}	8.3×10^{-4}
REGENERATIVE HEAT EXCHANGER	2.1×10^5	16	2400	-	9.4	<1	4.3	<1	8710	195	1.8	0.12	0.92	0.9	1.74	0.009	0.52
REACTOR SHUTDOWN COOLER (REGENERATIVE HEAT EXCHANGER)*	5.4×10^5	41	6200	-	24	<3	11	<3	22,360	500	4.5	0.30	2.37	2.41	4.47	0.023	1.33
FUEL PIT COOLER (FUEL POOL WALL)*	6.9×10^5	0.52	16	0.26	<0.2	18	180	0.10	152	2.2	0.07	0.59	0.008	0.009	0.052	0.002	0.034
SUPPRESSION CHAMBER	1.2×10^7	0.076	6.5	0.011	0.006	0.039	0.693	<0.002	20	0.39	0.11	<0.3	1.5×10^{-3}	1.9×10^{-3}	2.5×10^{-3}	6.1×10^{-5}	1.5×10^{-3}
SUPPRESSION COOLER (SUPPRESSION CHAMBER WALL)*	5.1×10^5	0.003	0.27	5×10^{-4}	<0.001	0.002	0.029	<0.001	0.84	0.016	0.005	<0.013	6.3×10^{-5}	8×10^{-5}	1.1×10^{-4}	2.6×10^{-6}	6.3×10^{-5}
CONDENSATE DEMINERALIZER (COND. DEMINERALIZER STRAINER)*	3.1×10^5	0.087	4.40	<0.004	0.081	<0.004	0.28	0.004	2540	0.315	0.008	<0.08	1.6×10^{-3}	1.9×10^{-3}	3.8×10^{-3}	84×10^{-6}	4.9×10^{-3}
WASTE RECOVERY TANKS (REGENERATIVE HEAT EXCHANGER)*	1.3×10^6	0.37	18.7	<0.020	0.34	<0.02	1.2	0.02	10,800	1.34	0.04	<0.33	6.6×10^{-3}	8.0×10^{-3}	1.6×10^{-2}	3.6×10^{-4}	20.9×10^{-3}
WASTE HOLD TANKS (REGENERATIVE HEAT EXCHANGER)*	8.8×10^5	0.25	12.4	<0.012	0.23	<0.011	0.79	0.012	7200	0.88	0.024	<0.22	4.4×10^{-3}	5.3×10^{-3}	1.1×10^{-2}	2.4×10^{-4}	13.9×10^{-3}
CONCENTRATION WASTE TANKS (REGENERATIVE HEAT EXCHANGER)*	8.7×10^5	0.24	12.2	<0.012	0.23	<0.011	0.78	0.011	7100	0.88	0.023	<0.22	4.3×10^{-3}	5.2×10^{-3}	1.0×10^{-2}	2.3×10^{-4}	13.7×10^{-3}
RESIN DISPENSER TANK (REGENERATIVE HEAT EXCHANGER)*	6.3×10^5	0.18	8.9	<0.009	0.16	<0.008	0.57	0.008	5190	0.64	0.017	<0.16	3.1×10^{-3}	3.8×10^{-3}	7.6×10^{-3}	1.7×10^{-4}	10.0×10^{-3}

*SAMPLE CONCENTRATIONS USED FOR BASIS OF INVENTORY ESTIMATE

TABLE C.2.17. Total Humboldt Bay Radionuclide Inventories(a)

Radionuclide	Half-Life (years)	1981 Inventory (millicuries)	Inventory at 1976 Shutdown (millicuries)
^{55}Fe	2.7	149,000	540,000
^{60}Co	5.27	18,000	35,000
^{137}Cs	30.2	2,200	2,500
^{63}Ni	100	1,400	1,400
^{54}Mn	0.855	336	19,000
^{90}Sr	28.5	17.9	18.1
^{241}AM	432	12.1	12
^{238}Pu	87.7	7.0	7
$^{239-240}$Pu	24,110	6.1	6
^{244}Cm	18.1	4.9	5.9
^{242}Cm	0.446	0.12	284
Total		171,000	596,000

(a) Excluding the pressure vessel, biological shield, concrete surfaces and residues in tanks and sumps.

C.3 RESIDUAL RADIONUCLIDE CONCENTRATIONS AND INVENTORIES AT DRESDEN UNIT ONE

TABLE C.3.1. Residual Radionuclide Concentrations in Corrosion Films from Dresden One - August 1982

	CONCENTRATION - pCi/cm^2 (pCi/gm)															
SAMPLE	^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho
D-SC-10 FUEL POOL WALL	1200 ± 100 (250 ± 30)	176,000 ± 1000 (35,800 ± 100)	1500 ± 300 (310 ± 60)	<40 (<8)	<700 (<100)	<500 <100)	<700 (<100)	<200 (<40)	<400 (<90)	10,600 ± 100 (2160 ± 20)	86,900 ± 200 (17,660 ± 40)	<300 (<60)	<100 (<30)	<200 (<30)	<100 (<20)	<60 (<10)
D-SC-11 FUEL POOL RECIRC. PUMP	<70	11,200 ± 200	<20	<21	<400	<300	<400	<100	<20	650 ± 50	7300 ± 100	<200	<80	<80	<60	<30
D-SC-12 FUEL POOL LADDER	1700 ± 100 (340 ± 30)	200,000 ± 1000 (41,600 ± 100)	2800 ± 300 (560 ± 60)	<40 (<7)	<600 (<100)	<400 (<80)	<600 (<100)	500 ± 200 (100 ± 40)	<30 (<7)	6000 ± 100 (1210 ± 10)	50,100 ± 200 (10,170 ± 20)	<200 (<50)	<100 (<20)	<100 (<30)	<80 (<20)	<50 (<10)
D-SC-13 INSIDE SPHERE WALL	<0.3	120 ± 1	2.7 ± 0.8	<0.1	<2	<1	<2	<0.4	<0.1	<0.3	3.4 ± 0.2	<0.6	<0.3	<0.4	<0.2	<0.1
D-SC-14 AIR VENT DUCT	<0.2	93 ± 1	1.2 ± 0.4	<0.1	<1	<0.6	<1	<0.2	<0.05	0.43 ± 0.09	5.5 ± 0.1	0.7 ± 0.4	<0.2	0.5 ± 0.2	<0.1	<0.08
D-SC-15 FLOOR DRAIN INSTR. ROOM A	34 ± 11 (15 ± 5)	26,400 ± 40 (11,630 ± 20)	390 ± 30 (170 ± 10)	<3 (<1)	<60 (<30)	<40 (<20)	<60 (<30)	<20 (<7)	<3 (<1)	990 ± 10 (437 ± 3)	7600 ± 30 (3360 ± 30)	<20 (<10)	<10 (<5)	<10 (<6)	<8 (<4)	<5 (<2)
D-SC-16 FUEL CHUTE	3.6 ± 0.9	790 ± 3	11 ± 2	<0.3	<4	<3	<5	<1	<0.2	3.3 ± 0.4	24 ± 1	4 ± 2	<0.8	<1	<0.5	<0.4
D-SC-17 FUEL CANAL	<0.2 (<0.04)	53 ± 1 (10.8 ± 0.2)	1.5 ± 0.5 (0.30 ± 0.10)	<0.1 (<0.01)	<1 (<0.2)	<1 (<0.2)	<1 (<0.2)	<0.4 (<0.08)	<0.08 (<0.02)	5.6 ± 0.2 (1.13 ± 0.04)	39 ± 1 (7.91 ± 0.07)	<0.5 (<0.1)	<0.3 (<0.05)	<0.3 (<0.06)	<0.2 (<0.04)	<0.1 (<0.02)
D-SC-18 REACTOR STEAM VENT LINE	12,000 ± 2000 (800 ± 100)	$7.90 \times 10^6 \pm 0.01 \times 10^6$ (522,000 ± 1000)	113,000 ± 4000 (7500 ± 300)	<500 (<30)	<8000 (<600)	<5000 (<300)	<9000 (<600)	<2000 (<100)	<400 (<30)	<800 (<50)	5500 ± 1000 (360 ± 60)	12,000 ± 3000 (800 ± 200)	<2000 (<100)	<2000 (<100)	<1000 (<70)	<700 (<50)
D-SC-19 STEAM RELIEF VALVE	<0.6	510 ± 2	7 ± 1	<0.2	<3	<2	<3	<0.7	<0.1	<0.3	<4	2 ± 1	<0.5	<0.7	<0.4	<0.3
D-SC-20 LOW PRESSURE INLET TO TURBINE	<2	2270 ± 10	37 ± 4	<0.5	<8	<5	<8	<4	<0.4	<0.7	3.1 ± 0.9	<3	5 ± 1	<2	<1	<0.7
D-SC-21 HIGH PRESS. OUTLET FROM TURBINE	<0.6	975 ± 2	15 ± 2	<0.2	<3	<2	<3	<0.7	<0.1	<0.3	6.6 ± 0.9	4 ± 1	<0.5	1.7 ± 0.7	<0.4	<0.3
D-SC-22 STEAM LINE PIPING	1 ± 0.5	2750 ± 10	36 ± 3	<0.3	<5	<3	12 ± 6	<1	<0.2	<0.5	2.0 ± 0.6	<2	2.4 ± 0.9	<1	<0.6	<0.4

TABLE C.3.2. Residual Radionuclide Concentrations in Corrosion Films from Dresden One, August 1982

SAMPLE	CONCENTRATION - pCi/cm² (pCi/gm) ^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I
D-SC-12 FUEL POOL LADDER	164,000 ± 10,000 (33,000 ± 2,000)	560 ± 40 (110 ± 10)	6,600 ± 500 (1,330 ± 90)	80 ± 5	45 ± 3 (9.1 ± 0.6)	<0.4 (<0.08)
D-SC-15 FLOOR DRAIN INSTRUMENT ROOM A	540 ± 30 (0.32 ± 0.02)	85 ± 5 (0.051 ± 0.003)	1,300 ± 80 (0.77 ± 0.05)	6,200 ± 400	0.93 ± 0.06 (0.0006 ± 0.0001)	0.09 ± 0.02 ($5 \times 10^{-5} \pm 1 \times 10^{-5}$)
D-SC-16 FUEL TRANSFER CHUTE	440 ± 30	8.9 ± 0.7	3,700 ± 300	0.062 ± 0.004	0.009 ± 0.002	<0.02
D-SC-18 REACTOR STEAM VENT LINE	2.91×10^6 (192,000 ± 11,000)	26,000 ± 2,000 (1,700 ± 100)	$1.28 \times 10^6 \pm 0.08 \times 10^6$ (85,000 ± 5,000)	690 ± 40	4.5 ± 0.3 (0.30 ± 0.02)	<1 (<0.07)
D-SC-20 LOW PRESSURE INLET TO TURBINE	2,200 ± 300	10 ± 1	1,000 ± 100	2.9 ± 0.2	0.05 ± 0.01	<0.01
D-SC-22 STEAM LINE PIPING	2,100 ± 200	10 ± 1	720 ± 90	3.8 ± 0.2	0.05 ± 0.02	0.03 ± 0.01

TABLE C.3.3. Residual Radionuclide Concentrations in Corrosion Films from Dresden One, August 1982

SAMPLE	CONCENTRATIONS - pCi/cm^2 (pCi/gm) ^{237}Np	^{238}Pu	^{239,240}Pu	^{241}Am	^{242}Cm	^{244}Cm
D-SC-12 FUEL POOL LADDER	0.034 ± 0.004 (0.007 ± 0.001)	61 ± 2 (12.5 ± 0.5)	24 ± 1 (4.8 ± 0.2)	33 ± 2 (6.7 ± 0.3)	7.2 ± 0.7 (1.5 ± 0.1)	56 ± 3 (11.4 ± 0.7)
D-SC-15 FLOOR DRAIN INSTRUMENT ROOM A	<0.003	7.3 ± 0.6	13.0 ± 0.7	20 ± 1	0.28 ± 0.03	1.7 ± 0.1
D-SC-16 FUEL TRANSFER CHUTE	0.0008 ± 0.0004	0.16 ± 0.01	0.067 ± 0.003	0.11 ± 0.01	0.020 ± 0.04	0.20 ± 0.01
D-SC-18 REACTOR STEAM VENT LINE	0.27 ± 0.13 (0.018 ± 0.009)	720 ± 30 (47 ± 2)	360 ± 10 (24 ± 1)	550 ± 30 (36 ± 2)	100 ± 10 (6.6 ± 0.6)	840 ± 50 (55 ± 3)
D-SC-20 LOW PRESSURE INLET TO TURBINE	0.0004 ± 0.0002	0.42 ± 0.02	0.68 ± 0.03	0.98 ± 0.04	0.015 ± 0.004	0.19 ± 0.01
D-SC-22 STEAM LINE PIPING	0.0006 ± 0.0003	0.21 ± 0.01	0.44 ± 0.02	0.70 ± 0.02	0.009 ± 0.003	0.17 ± 0.01

TABLE C.3.4. Radionuclide Concentrations* (pCi/cm^2) in Dresden Unit One Concrete Core Segments

	Location	Depth Interval (cm)	^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}As	^{125}Sb	^{126}Sn
1	Dow Chemical Spill 529' Elevation	0-1 1-2	<50 <20	208,800 ± 100 73,300 ± 400	<200 <60	<20 18 ± 7	<300 <100	<300 240 ± 70	<300 <80	<80 <20	<20 <20
2	Secondary Steam Generator Room "B" 529' Elevation	0-1 1-2	40 ± 20 0.5 ± 0.2	33,910 ± 50 320 ± 1	<70 <0.8	<8 <0.09	<200 4 ± 2	<200 <2	<90 <1	<60 <0.7	<20 <0.2
3	Hallway in Front of Accumulator Room488' 488' Elevation	0-1 1-2	27.8 ± 0.9 <0.03	7,010 ± 10 7.57 ± 0.09	<4 <0.04	<0.4 <0.02	34 ± 6 <0.3	<4 <0.3	<5 <0.2	1.5 ± 1.2 <0.2	<0.9 <0.02
4	Sub-pile Room 488' Elevation	0-1 1-2	960 ± 70 2.5 ± 0.2	276,900 ± 100 1,000 ± 1	<900 <2	<30 <0.2	<500 3 ± 1	<400 <2	<500 <0.7	<100 <0.6	<40 <0.2
5	Make-up Demineralizer 1-2	0-1	<0.2 <0.02	27.3 ± 0.3 1.98 ± 0.05	<0.4 <0.04	<0.05 <0.01	<1 <0.1	<2 <0.1	<0.5 <0.04	<0.5 <0.04	<0.1 <0.01
6	Condensate Pump Room Pit 1-2	0-1	54 ± 6 <0.06	19,050 ± 20 27.1 ± 0.2	<20 <0.2	<3 <0.03	<60 <0.5	<70 <0.4	<30 <0.3	<20 <0.1	<7 <0.04
7	Unloading Heat Exchanger Room 1-2	0-1 1.1 ± 0.2	310 ± 20 240.7 ± 0.5	108,200 ± 500 <0.7	<80 <0.08	<8 <1	<130 <0.7	180 ± 90 <0.9	350 ± 100 0.4 ± 0.2	<30 <0.06	<8
8	Rad Waste Basement 1-2	0-1 <0.05	29 ± 8 9.2 ± 0.1	7,560 ± 20 <0.2	<30 <0.02	<3 <1	<70 <2	<90 <0.2	<40 0.5 <0.1	<30	<8
9	Rad Waste Basement Drainage Trough	0-1 1-2	63 ± 8 0.10 ± 0.04	20,750 ± 20 11.1 ± 0.1	<80 <0.2	<4 <0.02	<80 2 ± 1	<90 <1	<40 <0.2	<30 0.6 ± 0.4	<8 <0.3

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* To convert to pCi/g, multiply by 0.45.

TABLE C.3.4. (continued) Radionuclide Concentrations* (pCi/cm^2) in Dresden Unite One Concrete Core Segments

Location	Internal (cm)	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166}Ho	^{228}Ac
1 Dow Chemical Spill 529' Elevatior	0-1	<30	130 ± 30	<260	<50	<40	<70	<30	<300
	1-2	<80	40 ± 10	<80	<7	<10	50 ± 20	<10	<80
2 Secondary Steam Generator Room "B" 529' Elevation	0-1	9,870 ± 30	202,200 ± 600	<300	<14	<13	<50	<11	<90
	1-2	102 ± 0.3	1,644 ± 4	4 ± 2	<0.4	<0.2	<0.4	<0.2	<1
3 Hallway in Front of Accumulator Room 486' Elevation	0-1	75.4 ± 0.8	948 ± 1	35 ± 3	<1.2	6.8 ± 0.9	2.4 ± 0.8	<0.6	<5
	1-2	0.72 ± 0.04	18.2 ± 0.1	<0.2	<0.06	<0.04	<0.05	<0.02	1.4 ± 0.2
4 Sub-pile Room 488' Elevation	0-1	39,970 ± 900	197,600 ± 400	800 ± 40	<80	<60	300 ± 100	<40	<400
	1-2	27.6 ± 0.2	1,203 ± 1	1.3 ± 0.7	13.1 ± 0.4	1.7 ± 0.3	<0.4	<0.2	<2
5 Make-up Demineralizer	0-1	8.9 ± 0.1	431.1 ± 0.8	<1	<0.1	<0.1	<0.3	<0.06	<0.9
	1-2	0.11 ± 0.02	1.13 ± 0.03	<0.07	<0.06	<0.04	<0.04	<0.02	0.8 ± 0.1
6 Condensate Pump Room Pit	0-1	15,100 ± 200	111,800 ± 400	180 ± 70	<40	<4	<20	<4	<30
	1-2	4.00 ± 0.09	44.6 ± 0.2	<0.4	0.11 ± 0.07	<0.06	<0.1	<0.05	<0.3
7 Unloading Heat Exchanger Room	0-1	640 ± 10	18,120 ± 20	160 ± 100	48 ± 12	49 ± 13	40 ± 30	<10	<100
	1-2	0.7 ± 0.1	11.2 ± 0.1	<0.6	<0.2	<0.1	<0.2	<0.2	<1
8 Rad Waste Basement	0-1	1,580 ± 10	40,780 ± 40	110 ± 80	<4	<5	<20	<5	<40
	1-2	18.9 ± 0.1	910 ± 1	<1	0.25 ± 0.08	<0.05	<0.3	<0.03	0.6 ± 0.2
9 Rad Waste Basement Drainage Trough	0-1	6,870 ± 10	90,500 ± 300	360 ± 80	<5	12 ± 6	<20	<6	<40
	1-2	18.2 ± 0.1	862 ± 3	<0.9	<0.06	<0.04	<0.2	<0.03	<0.2

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* To convert to pCi/g, multiply by 0.45.

TABLE C.3.5. Residual Radionuclide Concentrations in Dresden One Concrete Cores

CONCRETE CORE	CONCENTRATIONS pCi/cm^2*											
	^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I	^{237}Np	^{238}Pu	$^{239\,240}$Pu	^{241}Am	^{242}Cm	^{244}Cm
DCC-1 CHEMICAL SPILL	18,000 ± 1,000	2,000 ± 100	920,000 ± 80,000	3.0 ± 0.3	117 ± 7	0.8	0.036 ± 0.006	15 ± 1	12 ± 1	18 ± 1	0.49 ± 0.15	5.6 ± 0.4
DCC-4 SUB-PILE ROOM	54,000 ± 3,000	2,300 ± 200	770,000 ± 70,000	2,900 ± 200	2.4 ± 0.2	0.3	0.026 ± 0.005	48 ± 2	21 ± 1	30 ± 1	5.1 ± 0.5	45 ± 3
DCC-7 UNLOADING HEAT EXCHANGER ROOM	23,000 ± 1,000	670 ± 60	170,000 ± 20,000	17 ± 1	1.3 ± 0.1	0.3	0.014 ± 0.004	[illegible].7 ± 0.3	3.6 ± 0.2	6.3 ± 0.3	0.80 ± 0.18	8.2 ± 0.6

*TO CONVERT TO pCi/g MULTIPLY BY 0.45

TABLE C.3.6. Radionuclide Concentrations (pCi/g) in Selected Onsite Soils and Sediments from Dresden Nuclear Station (August 1982)

Location	^{40}K	^{54}Mn	^{60}Co	^{65}Zn	^{106}Ru	^{125}Sb	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu
Equipment Hatch Area North Side of Sphere	8.3 ± 0.2	0.02 ± 0.01	1.32 ± 0.02	<0.03	<0.07	<0.03	<0.01	0.49 ± 0.01	<0.04	<0.05	<0.04	<0.04
Equipment Storage Doorway East of Sphere	6.9 ± 0.2	0.05 ± 0.02	6.2 ± 0.1	<0.06	<0.1	<0.06	0.06 ± 0.01	1.93 ± 0.02	<0.07	<0.1	<0.08	<0.1
Refuel Building South of Sphere	7.7 ± 0.2	0.23 ± 0.02	13.8 ± 0.1	<0.06	0.2 ± 0.1	<0.06	0.27 ± 0.01	3.36 ± 0.02	0.17 ± 0.07	<0.09	<0.09	<0.03
Rad Waste Area In Front of Door 0-6"	7.4 ± 0.5	0.2 ± 0.1	55.9 ± 0.3	<0.3	<0.8	<0.3	1.71 ± 0.07	96.2 ± 0.3	<0.4	<0.6	<0.4	<0.2
6-12"	8.4 ± 0.5	<0.1	161 ± 1	<0.6	<1	<1	6.3 ± 0.1	260 ± 1	1.5 ± 0.6	<0.9	<0.7	<0.2
Sediments												
Inlet Canal 0-8"	14.4 ± 0.3	2.5 ± 0.9	3.1 ± 0.1	<9	9 ± 4	<0.2	<0.07	1.19 ± 0.02	<7	<0.1	<0.1	<0.1
Outlet Canal 0-8"	10.5 ± 0.3	<0.02	2.79 ± 0.03	<0.05	<1	<0.06	0.02 ± 0.01	2.46 ± 0.02	<0.07	<0.09	<0.07	<0.03

TABLE C.3.7. Residual Radionuclide Concentrations in Corrosion Films from Dresden One, August 1982

SAMPLE DESCRIPTION	SAMPLE #	CONCENTRATIONS · pCi/cm^2															
		^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho
FUEL POOL LADDER	D-SC-12	1,700	200,000	2,800	<40	<600	<400	<600	<500	<30	6,000	50,100	<200	<100	<100	<80	<50
FUEL TRANSFER CHUTE	D-SC-16	3.7	790	11	<0.3	<4	<3	<5	<1	<0.2	3.3	24	4	<0.8	<1	<0.5	<0.4
REACTOR STEAM VENT LINE	D-SC-18	12,000	7.9×10^6	113,000	<500	<8,000	<5,000	<9,000	<2,000	<400	<800	5,500	12,000	<2,000	<2,000	<1,000	<700
LOW PRESSURE INLET TO TURBINE	D-SC-20	<2	2,270	37	<0.5	<8	<5	<8	<4	<0.4	<0.7	3.1	<3	5	<2	<1	<0.7

SAMPLE DESCRIPTION	SAMPLE #	^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I	^{238}Pu	^{239,240}Pu	^{241}Am	^{242}Cm	^{244}Cm
FUEL POOL LADDER	D-SC-12	164,000	560	6,600	80	45	<0.4	61	24	33	7.2	56
FUEL TRANSFER CHUTE	D-SC-16	440	8.9	3,700	0.062	0.009	<0.02	0.16	0.067	0.11	0.020	0.20
REACTOR STEAM VENT LINE	D-SC-18	2.9×10^6	26,000	1.3×10^6	690	4.5	<1	720	360	550	100	840
LOW PRESSURE INLET TO TURBINE	D-SC-20	2,200	10	1,000	2.9	0.05	<0.01	0.42	0.68	0.93	0.015	0.19

TABLE C.3.8. Residual Radionuclide Inventories in Various Operating Systems of Dresden One, August 1982

	AREA (cm^2)	INVENTORY IN CURIES															
		^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho
(18) CONDENSATE	1.34×10^{7}	1.6×10^{1}	110	1.5	7×10^{3}	1×10^{1}	7×10^{2}	1×10^{1}	3×10^{2}	5×10^{3}	1×10^{2}	7.4×10^{2}	1.6×10^{1}	3×10^{2}	3×10^{2}	1×10^{2}	9×10^{3}
(18) FEED WATER	5.00×10^{7}	6.0×10^{1}	400	5.7	3×10^{2}	4×10^{1}	3×10^{1}	5×10^{1}	1×10^{1}	2×10^{2}	4×10^{2}	2.8×10^{1}	6.0×10^{1}	1×10^{1}	1×10^{1}	5×10^{2}	4×10^{2}
(20) STEAM	7.38×10^{7}	1×10^{4}	1.7×10^{1}	2.7×10^{3}	4×10^{5}	6×10^{4}	4×10^{4}	6×10^{4}	3×10^{4}	3×10^{5}	5×10^{5}	2.3×10^{4}	2×10^{4}	3.7×10^{4}	1×10^{4}	7×10^{5}	5×10^{5}
(18) UNLOADING HT EXCH	1.11×10^{6}	1.3×10^{2}	8.8	1.3×10^{1}	6×10^{4}	9×10^{3}	6×10^{3}	1×10^{2}	2×10^{3}	4×10^{4}	9×10^{4}	6.1×10^{3}	1.3×10^{2}	2×10^{4}	2×10^{4}	1×10^{4}	8×10^{5}
(18) REACTOR CLEANUP	3.25×10^{6}	3.9×10^{2}	26	3.7×10^{1}	2×10^{3}	3×10^{2}	2×10^{2}	3×10^{2}	7×10^{3}	1×10^{3}	3×10^{3}	1.8×10^{2}	3.8×10^{2}	7×10^{3}	3×10^{3}	3×10^{3}	2×10^{3}
(18) RAD WASTE	1.42×10^{7}	1.7×10^{1}	110	1.6	7×10^{3}	1×10^{1}	7×10^{2}	1×10^{1}	3×10^{2}	6×10^{3}	1×10^{2}	7.8×10^{2}	1.7×10^{1}	3×10^{2}	3×10^{2}	1×10^{2}	1×10^{3}
(12) FUEL STORAGE	1.00×10^{7}	1.7×10^{2}	2.0	2.8×10^{2}	4×10^{4}	6×10^{3}	4×10^{3}	6×10^{3}	5.0×10^{3}	3×10^{4}	6.0×10^{2}	5.0×10^{1}	2×10^{3}	1×10^{3}	1×10^{3}	8×10^{4}	5×10^{4}
(16) FUEL TRANSFER	7.55×10^{6}	2.8×10^{5}	6.0×10^{3}	8.3×10^{5}	2.2×10^{6}	3×10^{6}	2×10^{5}	4×10^{5}	8×10^{6}	2×10^{6}	2.5×10^{5}	1.8×10^{4}	3.0×10^{5}	6×10^{6}	8×10^{6}	4×10^{6}	3×10^{6}
TOTAL		1.0	660	9.3					5.0×10^{3}		0.06	0.96	0.98	3.7×10^{4}			

	^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I	^{238}Pu	^{239,240}Pu	^{241}Am	^{242}Cm	^{244}Cm
(18) CONDENSATE	40	3.5×10^{1}	17	8.9×10^{-3}	6.0×10^{5}	1×10^{5}	9.6×10^{3}	4.8×10^{3}	7.4×10^{3}	1.4×10^{3}	1.1×10^{2}
(18) FEED WATER	150	1.3	65	3.5×10^{-2}	2.3×10^{4}	5×10^{5}	3.6×10^{2}	1.8×10^{2}	2.8×10^{2}	5.1×10^{3}	4.2×10^{2}
(20) STEAM	1.6×10^{1}	7.4×10^{4}	7.4×10^{-2}	2.1×10^{-4}	3.7×10^{6}	7×10^{7}	3.1×10^{5}	5.0×10^{5}	7.2×10^{5}	1.1×10^{6}	1.4×10^{5}
(18) UNLOADING HT EXCH	3.3	2.9×10^{2}	1.4	7.4×10^{-4}	5.0×10^{6}	1×10^{6}	8.0×10^{4}	4.0×10^{4}	6.1×10^{4}	1.1×10^{4}	9.3×10^{4}
(18) REACTOR CLEANUP	9.7	8.5×10^{2}	4.2	2.2×10^{-3}	1.5×10^{5}	3×10^{6}	2.3×10^{3}	1.2×10^{3}	1.8×10^{3}	3.3×10^{4}	2.7×10^{3}
(18) RAD WASTE	42	3.7×10^{1}	18	8.8×10^{-3}	6.4×10^{5}	1×10^{5}	1.0×10^{2}	5.1×10^{3}	7.8×10^{3}	1.5×10^{3}	1.2×10^{2}
(12) FUEL STORAGE	1.6	5.6×10^{3}	6.6×10^{2}	8.0×10^{-4}	4.5×10^{4}	4×10^{6}	6.1×10^{4}	2.4×10^{4}	3.3×10^{4}	7.3×10^{5}	5.6×10^{4}
(16) FUEL TRANSFER	3.3×10^{3}	3.7×10^{5}	2.8×10^{-2}	4.7×10^{-7}	6.8×10^{8}	2×10^{7}	1.2×10^{6}	5.1×10^{7}	8.3×10^{7}	1.4×10^{7}	1.5×10^{6}
TOTAL	250	2.1	110	0.079	8.3×10^{4}		5.9×10^{2}	3.0×10^{2}	4.6×10^{2}	8.5×10^{3}	6.92×10 [illegible]

TABLE C.3.9. Total Residual Radionuclide Inventory at Dresden One August, 1982(a) and at Shutdown - October, 1978

Radionuclide (b)	Half-life (years)	Inventory (curies)	
		August, 1982	Oct. 31, 1978 (Shutdown)
^{60}Co	5.27	660	1,080
^{55}Fe	2.7	250	650
^{63}Ni	100	110	110
^{65}Zn	0.668	9.3	450
^{59}Ni	75,000	2.1	2.1
^{54}Mn	0.855	1.0	20
^{144}Ce	0.779	0.98	30
^{137}Cs	30.2	0.96	1.0
^{244}Cm	18	0.069	0.08
^{134}Cs	2.06	0.060	0.21
^{238}Pu	87.7	0.059	0.061
^{241}Am	432	0.046	0.046
$^{239-240}$Pu	24,100	0.030	0.030
^{242}Cm	0.446	0.0085	2.9
	Total	1,030	2,350

(a) Excluding neutron-activated pressure vessel and internals, biological shield, concrete surfaces, residues, sludges and resins in tanks and sumps, and spent fuel.

(b) Other long-lived radionuclides specifically listed in 10 CFR 61,(9) e.g., ^{94}Nb, ^{99}Tc, and ^{129}I, were not included in the inventory because of their insignificant concentrations in the residual radioactive corrosion films in the plant piping and equipment.

C.4 RESIDUAL RADIONUCLIDE CONCENTRATIONS AND INVENTORIES AT INDIAN POINT UNIT ONE

TABLE C.4.1. Direct Gamma-ray Analysis of Leached Hardware Samples from Indiant Point Station, Unit One

Sample ID	Surface Area (cm^2)	Concentration in pCi/cm^2 of Sampled Material								
		^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn
IPH-1	38.6	<1	15 ± 1	<2	<0.2	<5	<5	<5	<5	<0.3
IPH-2	---	---	---	---	---	---	---	---	---	---
IPH-3	248	<3	1180 ± 10	24 ± 7	<0.8	<30	<40	<20	<20	<3
IPH-4	---	---	---	---	---	---	---	---	---	---
IPH-5	929	<0.02	0.052 ± 0.002	<0.05	<0.006	<0.2	<0.005	<0.005	<0.02	<0.05
IPH-6	929	<0.02	0.158 ± 0.003	<0.05	<0.006	<0.1	<0.005	<0.04	<0.02	<0.01
IPH-7	929	<0.02	0.362 ± 0.005	<0.06	<0.007	<0.2	<0.01	<0.02	<0.05	<0.01
IPH-8	929	<0.01	0.057 ± 0.002	<0.04	<0.005	<0.1	<<0.005	<0.04	<0.02	<0.005
IPH-9	929	29.3 ± 0.4	138 ± 1	2.7 ± 0.6	<0.08	<3	<3	<2	<2	<0.2
IPH-10	26.9	<3	178 ± 7	<7	<0.7	<20	<20	<20	<15	<1
IPH-11	405	<0.7	321 ± 2	7.7 ± 1.7	<0.2	<5	13 ± 13	<5	<2	<0.2
IPH-12	86.1	460 ± 100	231100 ± 3500	3600 ± 200	<34	<600	<300	<600	3800 ± 200	<20
IPH-13	53.2	<4	157 ± 1	<5	<0.8	<20	<15	<10	80 ± 10	<1
IPH-14	137	<2	1180 ± 10	<5	<0.6	<20	<20	<10	<7	<1
IPH-15	40.5	<0.2	290 ± 2	<0.7	<0.1	<2	<5	<1	<2	<0.5
IPH-16	---	---	---	---	---	---	---	---	---	---
IPH-17	81.1	<1	102 ± 4	<5	<0.5	<10	<10	<10	<10	<1
IPH-18	122	<70	48200 ± 200	690 ± 170	<20	<600	<600	<400	<400	<50
IPH-19	155	486 ± 6	8900 ± 60	170 ± 10	<1	<40	<30	<3	<30	<3
IPH-20	527	<2	3640 ± 10	63 ± 4	<0.6	<10	<10	<10	<10	2.8 ± 1.1
IPH-21	58	<3	859 ± 5	17 ± 7	<0.7	30 ± 10	<9	<15	<3	<0.7
IPH-22	103	1500 ± 200	901000 ± 8000	540 ± 30	140 ± 60	<1000	<600	<1200	<400	<50
IPH-23	103	<10	7740 ± 20	170 ± 30	<3	<50	80 ± 20	<60	<20	<20
IPH-24	522	<0.8	634 ± 2	14 ± 2	<0.2	<8	<8	<4	<6	<0.6

TABLE C.4.1. (continued) Direct Gamma-ray Analysis of Leached Hardware Samples from Indian Point Station, Unit One

Sample ID	Surface Area (cm^2)	Concentration in pCi/cm^2 of Sampled Material							
		^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ac
IPH-1	38.6	<0.8	9.3 ± 0.8	<5	6 ± 2	<2	<1	<0.5	<3
IPH-2	---	---	---	---	---	---	---	---	---
IPH-3	248	1640 ± 10	21100 ± 40	<20	<4	<3	<8	<1	<10
IPH-4	---	---	---	---	---	---	---	---	---
IPH-5	929	<0.01	0.043 ± 0.022	<0.1	<0.02	<0.02	<0.01	<0.01	<0.05
IPH-6	929	<0.01	0.166 ± 0.003	<0.09	<0.02	<0.01	<0.01	<0.01	<0.05
IPH-7	929	0.186 ± 0.004	3.97 ± 0.02	<0.1	<0.02	<0.02	<0.02	<0.01	<0.08
IPH-8	929	<0.009	0.114 ± 0.002	<0.009	<0.02	<0.01	<0.01	<0.01	<0.04
IPH-9	929	100 ± 1	2040 ± 10	<2	<0.3	<0.3	<0.5	<0.1	<1
IPH-10	26.9	19 ± 4	360 ± 10	<10	<7	<7	<4	<1	<15
IPH-11	405	73.1 ± 0.7	850 ± 10	<2	<0.5	<2	<0.7	<0.2	<2
IPH-12	86.1	2710 ± 20	33900 ± 110	<200	<80	<100	<70	<50	<500
IPH-13	53.2	45.1 ± 1.9	720 ± 20	<9	<4	<9	<4	<0.9	<9
IPH-14	137	449 ± 6	15800 ± 70	<10	<2	<2	<4	<0.7	<7
IPH-15	40.5	152 ± 1	3390 ± 10	<2	<0.5	<0.2	<0.7	<0.2	<2
IPH-16	---	---	---	---	---	---	---	---	---
IPH-17	81.1	70 ± 2	1290 ± 10	<10	<2	<2	<4	<0.6	<7
IPH-18	122	20400 ± 80	330000 ± 1000	<300	<70	<70	<100	<20	<300
IPH-19	155	5970 ± 10	24600 ± 130	<20	<6	<6	<6	<2	<3
IPH-20	527	873 ± 2	20900 ± 200	<8	<2	<2	<2	<0.8	<8
IPH-21	58	24 ± 2	380 ± 10	<7	<3	<3	<2	<1	<10
IPH-22	103	1840 ± 100	22100 ± 100	<400	590 ± 170	<200	<100	<80	<1000
IPH-23	103	17 ± 8	280 ± 10	<20	<7	<10	<5	<4	<40
IPH-24	522	239 ± 2	4980 ± 40	<4	<0.8	<0.8	<1	<0.2	<4

TABLE C.4.2. Direct Gamma-ray Analysis of Leached Hardware Samples from Indian Point Station, Unit One

Sample ID	Mass (g)	Concentration in pCi/g of Sampled Material								
		^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn
IPH-1	126.3	<0.4	4.5 ± 0.4	<0.7	<0.08	<2	<2	<2	<2	<0.08
IPH-2	953.5	<0.5	457 ± 1	8.8 ± 1.4	<0.2	<6	<7	<3	<5	<0.5
IPH-3	350.2	<2	837 ± 3	17 ± 5	<0.6	<20	<30	<10	<17	<2
IPH-4	35.2	1100 ± 230	130000 ± 1000	1900 ± 500	<60	<2000	<3000	<1000	<2000	<200
IPH-5	2.46	<8	19.5 ± 0.8	<20	<2	<80	<2	<2	<8	<2
IPH-6	4.3	<5	34.2 ± 0.7	<10	<1	<20	<1	<9	<5	<2
IPH-7	2.48	<8	135 ± 2	<20	<3	<80	<4	<8	<20	<4
IPH-8	3.07	<3	17.3 ± 0.7	<10	<2	<30	<2	<10	<7	<2
IPH-9	---	---	---	---	---	---	---	---	---	---
IPH-10	75.3	<1	64 ± 3	<3	<0.3	<8	<7	<7	<5	<0.5
IPH-11	439.5	<0.7	296 ± 3	7.1 ± 1.6	<0.2	<5	1[illegible] ± 3	<5	<2	<0.2
IPH-12	549.2	70 ± 20	36200 ± 550	560 ± 40	<5	<90	<50	<90	600 ± 40	<4
IPH-13	193.4	<1	43.3 ± 0.4	<2	<0.2	<5	<4	<3	23 ± 4	<0.3
IPH-14	955.5	<0.3	170 ± 1	<0.7	<0.08	<3	<3	<2	<1	<0.2
IPH-15	---	---	---	---	---	---	---	---	---	---
IPH-16	200	<200	36900 ± 300	1700 ± 400	<50	<2500	<2500	<1000	<2000	<200
IPH-17	---	---	---	---	---	---	---	---	---	---
IPH-18	---	---	---	---	---	---	---	---	---	---
IPH-19	---	---	---	---	---	---	---	---	---	---
IPH-20	4402.1	<0.2	436 ± 1	7.5 ± 0.5	<0.07	<2	<2	<1	<1	0.3 ± 0.1
IPH-21	---	---	---	---	---	---	---	---	---	---
IPH-22	---	---	---	---	---	---	---	---	---	---
IPH-23	---	---	---	---	---	---	---	---	---	---
IPH-24	---	---	---	---	---	---	---	---	---	---

TABLE C.4.2. (continued) Direct Gamma-ray Analysis of Leached Hardware Samples from Indian Point Station, Unit One

Sample ID	Mass (g)	Concentration in pCi/g of Sampled Material							
		^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ac
IPH-1	126.3	<0.2	2.8 ± 0.2	<2	1.7 ± 0.7	<0.6	<0.4	<0.2	<0.8
IPH-2	953.5	440 ± 1	10100 ± 30	12 ± 4	<0.6	<0.6	<1	<0.2	<2
IPH-3	350.2	1160 ± 10	15000 ± 30	<20	<3	<2	<6	<0.9	<9
IPH-4	35.2	164000 ± 600	2090000 ± 3000	<1000	<300	<200	<600	<90	<850
IPH-5	2.46	<4	16 ± 8	<40	<8	<8	<4	<4	<20
IPH-6	4.3	<2	35.8 ± 0.7	<20	<5	<2	<2	<2	<10
IPH-7	2.48	70 ± 2	1490 ± 10	<40	<8	<8	<8	<4	<30
IPH-8	3.07	<3	34.5 ± 0.7	<30	<7	<3	<3	<3	<10
IPH-9	---	---	---	---	---	---	---	---	---
IPH-10	75.3	6.9 ± 1.5	130 ± 3	<4	<3	<3	<1	<0.5	<5
IPH-11	439.5	67.3 ± 0.6	780 ± 3	<2	<0.4	<2	<0.7	<0.2	<2
IPH-12	549.2	424 ± 4	5320 ± 20	<40	<10	<20	<10	<7	<70
IPH-13	193.4	12.4 ± 0.5	198 ± 5	<3	<1	<3	<1	<0.3	<3
IPH-14	955.5	64.4 ± 0.8	2260 ± 40	<2	<0.3	<0.3	<0.6	<0.1	<1
IPH-15	---	---	---	---	---	---	---	---	---
IPH-16	200	51000 ± 400	1570000 ± 2000	<2000	<200	<150	<450	<50	<500
IPH-17	---	---	---	---	---	---	---	---	---
IPH-18	---	---	---	---	---	---	---	---	---
IPH-19	---	---	---	---	---	---	---	---	---
IPH-20	4402.1	104 ± 1	2500 ± 30	<0.9	<0.2	<0.2	<0.2	<0.09	<0.9
IPH-21	---	---	---	---	---	---	---	---	---
IPH-22	---	---	---	---	---	---	---	---	---
IPH-23	---	---	---	---	---	---	---	---	---
IPH-24	---	---	---	---	---	---	---	---	---

TABLE C.4.3. Radiochemical Analysis of Corrosion Films on Collected Hardware and Scraping Samples Collected from Indian Point Station, Unit One

Sample ID	Sampled Area (cm^2)	Sampled Mass (g)	Concentration in pCi/cm^2 (pCi/g)				
			^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc
IP-H 4	---	35.2	--- (720 ± 80)	--- (3080 ± 250)	--- (370000 ± 50000)	--- (620 ± 40)	--- (1.9 ± 0.3)
IP-H 12	86.1	549.2	350000 ± 20000 (54000 ± 3000)	1500 ± 100 (235 ± 20)	205000 ± 20000 (32000 ± 3000)	47 ± 3 (7.4 ± 0.4)	7.1 ± 0.5 (1.11 ± 0.08)
IP-H 18	122	---	131000 ± 8000 (---)	690 ± 60 (---)	170000 ± 20000 (---)	54 ± 3 (---)	1.15 ± 0.9 (---)
IP-H 19	155	---	31000 ± 2000 (---)	71 ± 8 (---)	6900 ± 800 (---)	29 ± 2 (---)	0.79 ±0.05 (---)
IP-H 20	527	4402	230 ± 10 (28 ± 2)	15 ± 1 (1.8 ± 0.1)	2600 ± 200 (310 ± 20)	3.1 ± 0.2 (0.37 ± 0.02)	0.21 ±0.01 (0.025± 0.002)
IP-H 22	103	---	1610000 ± 100000 (---)	2760 ± 140 (---)	318000 ± 20000 (---)	96 ± 6 (---)	14.3 ± 0.7 (---)

TABLE C.4.3 (continued) Radiochemical Analysis of Corrosion Films on Collected Hardware and Scraping Samples Collected from Indian Point Station, Unit One

Sample ID	Sampled Area (cm^2)	Sampled Mass (g)	Concentration pCi/cm^2 (pCi/g)					
			^{129}I	^{237}Np	^{238}Pu	$^{239-240}Pu$	^{241}Am	^{244}Cm
IP-H 4	---	35.2	--- (<0.6)	--- (<0.008)	--- (0.049 ± 0.004)	--- (0.049 ± 0.004)	--- (0.102 ± 0.008)	--- (0.026 ± 0.004)
IP-H 12	86.1	549.2	<0.4 (<0.6)	<0.03 (<0.005)	7.5 ± 0.3 (1.18 ± 0.05)	13.5 ± 0.5 (2.12 ± 0.08)	21.7 ± 0.9 (0.0034 ± 0.0001)	2.8 ± 0.1 (0.440 ± 0.004)
IP-H 18	122	---	0.39 ± 0.11 (---)	0.0054 ± 0.0012 (---)	4.4 ± 0.2 (---)	8.5 ± 0.3 (---)	9.8 ± 0.3 (---)	0.95 ± 0.06) (---)
IP-H 19	155	---	0.21 ± 0.12 (---)	0.012 ± 0.002 (---)	0.129 ± 0.006 (---)	0.163 ± 0.006 (---)	0.155 ± 0.008 (---)	0.069 ± 0.006 (---)
IP-H 20	527	4402	0.056 ± 0.017 (0.0068 ± 0.0020)	<0.0003 (<0.00004)	0.156 ± 0.008 (0.0187 ± 0.0009)	0.32 ± 0.02 (0.038 ± 0.002)	0.87 ± 0.03 (0.105 ± 0.003)	0.054 ± 0.004 (0.0065 ± 0.0005)
IP-H 22	103	---	4.3 ± 0.3 (---)	<0.01 (---)	44 ± 2 (---)	77 ± 3 (---)	102 ± 7 (---)	12.3 ± 0.9 (---)

TABLE C.4.4. Radionuclide Concentration in Concrete Core Segments from Indian Point Station, Unit One

Sample ID	Core Depth (cm)	Sample Weight (g)	Concentration in pCi/cm^2*								
			^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn
IP-CC 1	0-1	114.7	4.8 ± 0.4	625 ± 1	<2	<0.1	<5	<5	<2	<3	<0.4
	1-2	120.7	<0.03	1.46 ± 0.05	<0.1	<0.01	<0.3	<0.2	<0.1	<0.06	<0.02
IP-CC 2	0-1	116.2	0.58 ± 0.26	464 ± 1	<1	<0.1	<3	<3	<2	<2	<0.3
	1-2	155.6	<0.2	52.7 ± 0.4	<0.7	<0.06	<2	<2	<0.9	<1	<0.2
IP-CC 3	0-1	134.5	13.2 ± 0.7	928 ± 1	<3	<0.3	<9	<9	<3	<1	<0.8
	1-2	116.5	0.81 ± 0.20	65.7 ± 0.4	<0.8	<0.07	<2	<2	<1	<0.8	<0.2
IP-CC 4	0-1	135.6	<2	938 ± 5	<9	2.3 ± 0.9	<40	<50	<11	<40	<4
	1-2	199	0.35 ± 0.11	51.2 ± 0.2	<0.5	<0.04	<1	<1	<0.6	<0.8	<0.1
IP-CC 5	0-1[a]	31	<800	15400 ± 300	<2000	<400	<9000	<8000	<3000	<6000	<700
	1-2.5	246.1	0.72 ± 0.16	76.8 ± 0.3	1.4	<0.07	<3	<3	<0.8	<2	<0.2
IP-CC 6	0-1	114.9	80 ± 12	4820 ± 10	<30	<6	<200	<200	<40	<150	<20
	1-2	141.2	<0.08	6.8 ± 0.1	<0.3	<0.04	<2	<2	<0.4	<1	<0.2
	2-3	211.8	<0.07	9.0 ± 0.1	<0.3	<0.03	<2	<2	<0.3	<1	<0.2
	3-4	119.7	<0.04	2.84 ± 0.06	<0.1	<0.02	<0.7	<0.7	<0.2	<0.5	<0.06
	4-8	106.2	<0.1	11.7 ± 0.2	<0.5	<0.05	<2	<2	<0.6	<1	<0.2
IP-CC 7	0-1	145.6	121 ± 15	5390 ± 20	<40	<8	<200	<300	<60	<90	<20
	1-2	172.4	0.17 ± 0.07	15.0 ± 0.1	<0.3	<0.03	<1	<1	<0.4	<0.3	<0.08
	2-3	125.6	<0.04	3.02 ± 0.07	<0.2	<0.02	<0.5	<0.04	<0.2	<0.03	<0.04
	3-5	279.8	0.58 ± 0.20	51.8 ± 0.4	<0.9	<0.07	<3	<2	<1	<2	<0.2
	5-6	162.4	0.15 ± 0.06	1.75 ± 0.09	<0.3	<0.03	<1	<1	<0.3	<0.3	<0.08
	6-7	144.8	<0.04	1.46 ± 0.05	<0.2	<0.02	<0.6	<0.6	<0.2	<0.2	<0.05
	7-8	80.2	<0.04	1.14 ± 0.05	<0.1	<0.01	<0.5	1.3 ± 0.4	<0.1	<0.3	<0.04
	8-9	136.7	<0.06	1.49 ± 0.08	<0.2	<0.03	<1	<1	<0.2	<0.8	<0.09
	9-10	75.8	<0.04	0.70 ± 0.05	<0.1	<0.02	<0.6	<0.6	<0.2	<0.4	<0.05
	10-8	430.5	<0.09	3.1 ± 0.1	<0.4	<0.04	<2	<2	<0.4	<1	<0.1

* To obtain pCi/g, multiply by 71.0 cm^2 and divide by the weight

[a] Use an area of 13.54 cm^2 when determining pCi/g

TABLE C.4.4. (continued) Radionuclide Concentration in Concrete Core Segments from Indian Point Station, Unit One

Sample ID	Core Depth (cm)	Sample Weight (g)	Concentration in pCi/cm²*							
			^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ac
IP-CC 1	0-1	114.7	365 ± 1	10600 ± 30	<4	<0.6	<0.3	<0.9	<0.2	<2
	1-2	120.7	<0.02	3.55 ± 0.05	<0.2	<0.07	<0.04	<0.05	<0.02	0.86 ± 0.15
IP-CC 2	0-1	116.2	181 ± 1	4600 ± 2	<2	<0.4	<0.2	<0.5	<0.2	<1
	1-2	155.6	17.7 ± 0.2	706 ± 1	<2	<0.2	<0.2	<0.4	<0.1	<0.8
IP-CC 3	0-1	134.5	1610 ± 10	26120 ± 40	<8	<1	<0.5	<2	<0.4	<3
	1-2	116.5	48.0 ± 0.4	736 ± 1	<2	<0.2	<0.1	<0.4	<0.09	<0.8
IP-CC 4	0-1	135.6	1160 ± 10	38900 ± 200	<60	<1	<1	<28 ± 13	<1	<10
	1-2	199	10.3 ± 0.1	521 ± 1	<1	<0.1	<0.7	<0.2	<0.05	<0.5
IP-CC 5	0-1[a]	31	627000 ± 1000	230000 ± 8000	24000 ± 9000	<500	<400	<2000	<500	<2000
	1-2.5	246.1	228 ± 1	2760 ± 10	6.7 ± 3.0	<0.7	<0.1	<0.6	<0.09	<0.6
IP-CC 6	0-1	114.9	37000 ± 300	583000 ± 1000	970 ± 100	<8	<7	<20	<8	<40
	1-2	141.2	52.4 ± 0.3	638 ± 1	<2	<0.1	<0.07	<0.4	<0.05	1.7 ± 0.3
	2-3	211.8	80.8 ± 0.3	1028 ± 1	<2	<0.1	<0.06	<0.4	<0.04	1.8 ± 0.3
	3-4	119.7	15.4 ± 0.1	198 ± 1	<0.6	<0.06	<0.04	<0.1	<0.02	1.4 ± 0.2
	4-B	106.2	38.9 ± 0.4	511 ± 1	<2	<0.2	<0.1	<0.4	<0.07	2.1 ± 0.5
IP-CC 7	0-1	145.6	30640 ± 40	433000 ± 800	1510 ± 310	<10	<9	<70	<10	<50
	1-2	172.4	17.0 ± 0.1	248 ± 1	<0.9	<0.1	<0.06	<0.2	<0.04	1.6 ± 0.3
	2-3	125.6	3.98 ± 0.06	54.0 ± 0.2	<0.04	<0.08	<0.04	<0.09	<0.02	1.2 ± 0.2
	3-5	279.8	56.2 ± 0.4	725 ± 1	<3	<0.2	<0.2	<0.5	<0.1	<0.8
	5-6	162.4	9.1 ± 0.2	107 ± 1	<1	<0.1	<0.07	<0.2	<0.03	1.5 ± 0.3
	6-7	144.8	7.73 ± 0.09	94.3 ± 0.3	<0.6	<0.07	<0.05	<0.1	<0.02	1.7 ± 0.2
	7-8	80.2	5.19 ± 0.07	61.8 ± 0.2	<0.5	<0.07	<0.04	<0.09	<0.02	1.2 ± 0.1
	8-9	136.7	12.1 ± 0.2	140 ± 1	<1	<0.1	<0.07	<0.2	<0.03	1.5 ± 0.2
	9-10	75.8	7.76 ± 0.09	88.7 ± 0.3	<0.6	<0.8	<0.05	<0.1	<0.02	1.5 ± 0.2
	10-B	430.5	22.2 ± 0.2	252 ± 1	<2	<0.2	<0.1	<0.4	<0.05	3.7 ± 0.4

* To obtain pCi/g, multiply by 71.0 cm² and divide by the weight

[a] Use an area of 13.54 cm² when determining pCi/g

TABLE C.4.5. Radionuclide Concentration in Concrete Core Segments from Indian Point Station, Unit One

Sample ID	Core Depth (cm)	Sample Weight (g)	Concentration in pCi/cm²*								
			^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn
IP-CC 8	0-1	129.4	<30	1320 ± 60	<100	<20	<700	<900	<200	<700	<80
	1-2	145.8	<0.04	3.69 ± 0.07	<0.2	<0.02	<0.4	<0.3	<0.2	<0.2	<0.03
IP-CC 9	0-1[a]	71.1	<100	12600 ± 20	<200	<50	<1300	<1400	<400	<1000	<120
	1-2	128.2	<0.07	10.7 ± 0.1	<0.3	<0.03	<1	<1	<0.3	<1	<0.1
	2-3	112.5	<0.06	6.4 ± 0.1	<0.3	<0.03	<1	<1	<0.3	<0.8	<0.1
	3-4	132.1	<0.07	5.29 ± 0.09	<0.3	<0.03	<2	<2	<0.3	<1	<0.2
	4-8	415.5	<0.4	76.8 ± 0.5	<1	<0.2	<8	<8	<2	<6	<0.7
IP-CC 10	0-1	159.1	14.1 ± 0.5	493 ± 1	<2	<0.2	<7	<6	<2	<4	<0.6
	1-2	125.2	<0.03	1.60 ± 0.05	<0.1	<0.01	<0.3	<0.2	<0.2	<0.2	<0.02
IP-CC 11	0-1	144.4	314 ± 13	10110 ± 30	<40	<5	<200	<200	<60	<150	<20
	1-2	150.8	3.4 ± 0.5	478 ± 2	<2	<0.2	<3	<2	<3	<1	<0.2
IP-CC 12	0-1	172.3	12 ± 4	1590 ± 10	<14	<1	<60	<80	<20	<60	18 ± 7
	1-2	169.9	6.1 ± 0.5	905 ± 1	<2	<0.2	<5	<4	<3	<3	<0.3
IP-CC 13	0-1	128.6	<0.1	36.4 ± 0.3	<0.6	<0.05	<1	<0.9	<0.8	<0.6	<0.08
	1-2	187	<0.03	0.34 ± 0.04	<0.1	<0.01	<0.2	<0.2	<0.1	<0.2	<0.01
IP-CC 14	0-1	132.4	74 ± 4	1090 ± 10	<12	<2	<70	<90	<16	<70	<8
	1-2	150.9	<0.4	286 ± 1	<2	<0.1	<3	<2	<2	<2	<0.2
IP-CC 15	0-1[b]	66.5	<200	11300 ± 100	<500	<100	<3000	<3000	<800	<2000	<200
	1-2	162.1	<0.1	71.2 ± 0.3	<0.6	<0.04	<0.9	<0.6	<0.7	<0.5	<0.05
	2-3	127.1	<0.1	2.9 ± 0.3	<0.6	<0.06	<1	<1	<0.6	<1	<0.1
	3-4	146.9	<0.04	1.98 ± 0.06	<0.2	<0.02	<0.4	<0.4	<0.2	<0.3	<0.03
	4-8	541.1	<0.03	7.31 ± 0.09	<0.09	<0.02	<0.5	<0.9	<0.1	<0.6	<0.06

* To obtain pCi/g, multiply by 71.0 cm² and divide by the weight

[a] Use an area of 33.9 cm² when determining pCi/g

[b] Use an area of 37.7 cm² when determining pCi/g

TABLE C.4.5 (continued) Radionuclide Concentration in Concrete Core Segments from Indian Point Station, Unit One

Sample ID	Core Depth (cm)	Sample Weight (g)	Concentration in pCi/cm²*							
			^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	[illegible]	^{228}Ac
IP-CC 8	0-1	129.4	6670 ± 130	123000 ± 100	<1000	<20	<20	<230	<20	<100
	1-2	145.8	1.40 ± 0.05	20.2 ± 0.4	<0.3	0.16 ± 0.08	<0.05	0.17 ± 0.07	<0.02	1.2 ± 0.2
IP-CC 9	0-1ᵃ	71.1	136000 ± 200	1640000 ± 3000	3700 ± 1900	<40	<50	<300	<70	<300
	1-2	128.2	55.8 ± 0.2	770 ± 1	<1	<0.1	<0.06	<0.3	<0.04	1.1 ± 0.3
	2-3	112.5	25.0 ± 0.2	312 ± 1	<1	<0.1	<0.05	<0.2	<0.08	1.2 ± 0.3
	3-4	132.1	92.2 ± 0.3	1110 ± 10	<2	<0.1	<0.08	<0.4	<0.05	0.99 ± 0.27
	4-8	415.4	662 ± 1	8070 ± 20	<8	<0.4	<0.2	<2	<0.2	<1
IP-CC 10	0-1	159.1	2340 ± 20	14000 ± 100	<6	<1	<0.4	<1	<0.3	<2
	1-2	125.2	0.85 ± 0.08	6.13 ± 0.06	<0.3	<0.07	<0.04	<0.06	<0.02	1.3 ± 0.1
IP-CC 11	0-1	144.4	20120 ± 30	223000 ± 600	<200	<8	<7	<90	<7	<50
	1-2	150.8	10.0 ± 0.4	223 ± 1	<2	<0.5	<0.4	<0.4	<0.2	<2
IP-CC 12	0-1	172.3	2290 ± 10	68200 ± 300	<100	<2	<2	<20	<2	<15
	1-2	169.9	124 ± 1	4130 ± 10	<4	<0.5	<0.4	<0.8	<0.2	<2
IP-CC 13	0-1	128.6	5.1 ± 0.2	97.2 ± 0.4	<0.9	<0.2	<0.1	<0.2	<0.07	<0.6
	1-2	187	<0.7	2.71 ± 0.05	<0.2	<0.2	<0.04	<0.06	<0.02	1.3 ± 0.1
IP-CC 14	0-1	132.4	2700 ± 10	88300 ± 400	270 ± 100	<2	<2	<20	<2	<10
	1-2	150.9	16.7 ± 0.3	670 ± 1	<2	<0.3	<0.2	<0.5	<0.2	<2
IP-CC 15	0-1ᵇ	66.5	106000 ± 100	2050000 ± 4000	7600 ± 3000	<40	<30	<700	<200	<600
	1-2	162.1	3.65 ± 0.08	70.3 ± 0.2	<0.6	<0.1	<0.1	<0.1	<0.06	<0.6
	2-3	127.1	1.6 ± 0.2	30.0 ± 0.6	<1	<0.2	<0.2	<0.3	<0.08	<0.7
	3-4	146.9	2.52 ± 0.06	47.2 ± 0.2	<0.4	0.13 ± 0.06	<0.04	<0.09	<0.02	1.4 ± 0.2
	4-8	541.1	9.23 ± 0.09	236 ± 1	<0.4	<0.08	<0.05	<0.2	<0.03	1.9 ± 0.2

* To obtain pCi/g, multiply by 71.0 cm² and divide by the weight

[a] Use an area of 33.9 cm² when determining pCi/g

[b] Use an area of 37.7 cm² when determining pCi/g

TABLE C.4.6. Radionuclide Analysis of Leachate Solutions for the Concrete Cores

Sample I.D.	Sampled Area (cm^2)	Sample Mass (g)	Concentration in pCi/cm^2 (pCi/g)								
			^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn
IP-CC-3 (0-1 cm)	90.5	67.3	2.8 ± 1.0 (13.7 ± 1.3)	234 ± 2 (315 ± 3)	<2 (<3)	<0.3 (<0.4)	<10 (<10)	<10 (<20)	<6 (<7)	<6 (<7)	<1 (<1)
IP-CC-9 (0-1 cm)	93.2	74.6	99 ± 31 (120 ± 40)	7330 ± 40 (9160 ± 50)	200 ± 60 (250 ± 80)	<10 (<10)	<500 (<700)	<800 (<900)	<100 (<200)	<400 (<500)	<50 (<70)
IP-CC-15 (0-1 cm)	80.4	55.8	<50 (<70)	7540 ± 70 (10900 ± 110)	<100 (<140)	<10 (<20)	<1000 (<1000)	<1200 (<2000)	<200 (<400)	<900 (<1000)	<100 (<100)

Sample I.D.	Sampled Area (cm^2)	Sample Mass (g)	Concentration in pCi/cm^2 (pci/g)							
			^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ac
IP-CC-3 (0-1 cm)	90.5	67.3	309 ± 8 (414 ± 10)	4590 ± 10 (6170 ± 10)	<8 (<10)	<1 (<1)	<1 (<1)	<2 (<3)	<0.4 (<0.6)	<4 (<6)
IP-CC-9 (0-1 cm)	93.2	74.6	82800 ± 100 (103000 ± 100)	908000 ± 1100 (1220000 ± 1000)	<400 (<500)	<80 (<90)	<30 (<40)	<100 (<150)	<10 (<10)	<100 (<100)
IP-CC-15 (0-1 cm)	80.4	55.8	49100 ± 200 (70000 ± 400)	943000 ± 1200 (1380000 ± 2000)	<600 (<900)	<100 (<180)	[illegible] (<25)	<200 (<400)	<30 (<40)	<200 (<400)

TABLE C.4.7. Radiochemical Analysis of Selected Concrete Core Segments at Indian Point Station, Unit One

Sample ID	Sampled Area (cm^2)	Sampled Mass (g)	Concentration in pCi/cm^2 (pCi/g)					
			^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I
IP-CC 3 (0-1 cm)	90.5	67.3	<2 (<4)	-1 ± 1 (-2 ± 2)	30 ± 3 (41 ± 4)	2.1 ± 0.2 (2.9 ± 0.2)	0.21 ± 0.04 (0.28 ± 0.06)	0.13 ± 0.04 (0.18 ± 0.05)
IP-CC 9 (0-1 cm)	93.2	74.6	315 ± 30 (390 ± 40)	30 ± 3 (37 ± 4)	2980 ± 330 (3700 ± 400)	46 ± 3 (58 ± 3)	0.27 ± 0.02 (0.34 ± 0.03)	0.17 ± 0.08 (0.2 ± 0.1)
IP-CC 15 (0-1 cm)	80.4	55.8	1060 ± 70 (1530 ± 110)	38 ± 4 (55 ± 6)	2240 ± 220 (3220 ± 320)	19 ± 1 (27 ± 2)	<0.09 (<0.1)	0.27 ± 0.11 (0.39 ± 0.16

Sample ID	Sampled Area (cm^2)	Sampled Mass (g)	Concentration in pCi/cm^2 (pCi/g)				
			^{237}Np	^{238}Pu	$^{239-240}$Pu	^{241}Am	^{244}Cm
IP-CC 3 (0-1 cm)	90.5	67.3	0.0035 ± 0.0014 (0.005 ± 0.002)	0.014 ± 0.002 (0.019 ± 0.002)	0.022 ± 0.002 (0.030 ± 0.003)	0.051 ± 0.011 (0.069 ± 0.015)	<0.02 (<0.02)
IP-CC 9 (0-1 cm)	93.2	74.6	0.013 ± 0.002 (0.016 ± 0.002)	0.051 ± 0.004 (0.064 ± 0.004)	0.089 ± 0.005 (0.112 ± 0.007)	0.13 ± 0.01 (0.17 ± 0.01)	0.09 ± 0.01 (0.11 ± 0.01)
IP-CC 15 (0-1 cm)	80.4	55.8	<0.005 (<0.008	0.020 ± 0.002 (0.029 ± 0.004)	0.023 ± 0.002 (0.033 ± 0.004)	0.043 ± 0.008 (0.06 ± 0.01)	0.014 ± 0.005 (0.020 ± 0.007)

TABLE C.4.8. Samples Used for Radionuclide Inventory

System	Sample Used for Inventory Estimate
A. Primary Coolant	IP-H22
B. Seal water and primary makeup	IP-H3, IP-H14
C. Main steam line	IP-H11, IP-H15
D. Condensate and condensate return	IP-H11, IP-H15
E. Spent fuel facility	IP-H21
F. Radwaste	IP-H18, IP-H20

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TABLE C.4.9. Residual Radionuclide Concentrations Used for Calculation of Radionuclide Inventory at Indian Point Station, Unit One, May 1982

	CONCENTRATION IN pCi/cm^2													
	^{54}Mn	^{60}Co	^{65}Zn	^{95}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu
A. PRIMARY COOLANT														
IP-H22	1500 ± 200	901 ± 8	540 ± 30	140 ± 60	<1000	<800	<1200	<400	<50	1840 ± 100	22.100 ± 100	<400	590 ± 170	<200
B. SEAL WATER AND PRIMARY MAKEUP														
IP-H3	<3	1.18 ± 0.01	24 ± 7	<0.8	<30	<40	<20	<20	<3	1640 ± 10	21.100 ± 100	<20	<4	<3
IP-H14	<2	1.18 ± 0.01	<5	<0.6	<20	<20	<10	<7	<1	448 ± 6	15.800 ± 100	<100	<2	<2
AVG	<2	1.18	24	<0.7	<25	<30	<15	<13	<2	1044	18.400	<15	<3	<2
C. MAIN STEAM LINE														
IP-H11	<0.7	0.321 ± 0.002	7.7 ± 1.7	<0.2	<5	13 ± 3	<5	<2	<0.2	73.1 ± 0.7	850 ± 100	<2	<0.5	<2
IP-H115	<0.2	0.290 ± 0.002	<0.7	<0.1	<2	<5	<1	<2	<0.5	152 ± 1	3390 ± 10	<2	<0.5	<0.2
AVG	<0.4	0.306	7.7	<0.1	<3	13	<3	<2	<0.3	112	2100	<2	<0.5	<1
D. CONDENSER AND CONDENSATE RETURN														
(USE AVG. FROM C)	<0.4	0.306	7.7	<0.1	<3	13	<3	<2	<0.3	112	2100	<2	<0.5	<1
E. FUEL STORAGE AREA														
IP-H21	<3	0.859 ± 0.005	17 ± 7	<0.7	30 ± 10	<9	<15	<3	<0.7	24 ± 2	380 ± 10	<1	<3	<3
F. LIQUID RADWASTE														
IP-H18	<70	48.2 ± 0.2	690 ± 170	<20	<600	<600	<400	<400	<50	20.400 ± 80	330,000 ± 1000	<300	<70	<70
IP-H20	<2	364.0 ± 0.1	63 ± 4	<0.6	<10	<10	<10	<10	28 ± 1.1	873 ± 2	20,900 ± 200	<8	<2	<2
AVG.	<36	206	376	<10.3	<305	<305	<205	<205	2.8	10.600	175,000	<150	<36	<36

TABLE C.4.9. (continued) Residual Radionuclide Concentrations Used for Calculation of Radionuclide Inventory at Indian Point Station, Unit One, May 1982

CONCENTRATION IN pCi/cm^2 UNLESS NOTED

	^{152}Eu	^{166m}Ho	^{226}Ra	^{60}Co	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I	^{237}Np	^{238}Pu	^{239,240}Pu	^{241}Am	^{244}Cm
A PRIMARY COOLANT														
IP-H22	<100	<80	<1000	1,818,000 ± 100	2780 ± 140	320,000 ± 20,000	98 ± 6	14.3 ± 0.7	4.3 ± 0.3	<0.01	44 ± 2	77 ± 3	102 ± 7	12.3 ± 0.9
B SEAL WATER AND PRIMARY MAKEUP														
IP-H3	<8	<1	<10											
IP-H14	<4	<0.7	<7											
AVG	<6	<0.8	<8	2100*	3.8*	420*	0.13*	0.019*	0.0056*	<0.00001*	0.068*	0.10*	0.14*	0.018*
C MAIN STEAM LINE														
IP-H11	<0.7	<0.2	<2											
IP-H15	<0.7	<0.2	<2											
AVG	<[illegible]	<0.2	<2	550*	0.9*	110*	0.033*	0.005*	0.0015*	<0.000003*	0.015*	0.03*	0.03*	0.004*
D CONDENSER AND CONDENSATE RETURN														
(USE AVG. FROM C)	<0.7	<0.2	<2	550*	0.9*	110*	0.033*	0.005*	0.0015*	<0.000003*	0.015*	0.03*	0.03*	0.004*
E FUEL STORAGE AREA														
IP-H21	<2	<1	<10	1600*	2.8*	305*	0.092*	0.014*	0.0041*	<0.000009	0.042*	0.07*	0.10*	0.012*
F LIQUID RADWASTE														
IP-H19	<100	<20	<300	131,000 ± 8000	690 ± 90	170,000 ± 20,000	54.3	1.16 ± 0.08	0.39 ± 0.11	0.0054 ± 0.0012	4.4 ± 0.2	8.5 ± 0.3	9.8 ± 0.3	0.95 ± 0.06
IP-H20	<2	<1	<10	220 ± 10	15 ± 1	2800 ± 200	3.1 ± 0.2	0.21 ± 0.01	0.056 ± 0.017	<0.0003	0.198 ± 0.006	0.32 ± 0.02	0.87 ± 0.03	0.064 ± 0.004
AVG	<51	<10	<150	66,000	352	86,000	28	0.68	0.22	0.0054	2.3	4.4	5.3	0.50

*VALUE CALCULATED BY RATIOING ELEMENT CONCENTRATION TO ^{60}Co IN PRIMARY COOLANT (IP-H22) AND MULTIPLYING BY ^{60}Co CONCENTRATION IN SAMPLE AVERAGE

TABLE C.4.10. Radionuclide Inventory for Indian Point Station, Unit One, May 1982

	AREA (cm²)	TOTAL ACTIVITY IN CURIES													
		[illegible]	^{60}Co	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	^{134}Cs	^{137}Cs	^{144}Ce	[illegible]	[illegible]
A. PRIMARY COOLANT	8.25×10^7	0.024	38.3	0.034	0.020	<0.09	<0.03	<3.08	<0.02	<0.003	0.17	0.8014	<0.02	0.037	<0.01
B. SEAL WATER AND PRIMARY MAKEUP	8.99×10^5	$<1 \times 10^{-5}$	7.8×10^{-3}	1.5×10^{-4}	$<4 \times 10^{-5}$	$<2 \times 10^{-4}$	$<2 \times 10^{-4}$	$<1 \times 10^{-4}$	$<8 \times 10^{-5}$	$<1 \times 10^{-5}$	6.9×10^{-3}	0.12	$<1 \times 10^{-4}$	$<2 \times 10^{-5}$	$<1 \times 10^{-5}$
C. MAIN STEAM LINE	1.62×10^6	$<5 \times 10^{-5}$	5.0×10^{-2}	1.2×10^{-3}	$<2 \times 10^{-5}$	$<5 \times 10^{-4}$	2.1×10^{-3}	$<5 \times 10^{-4}$	$<3 \times 10^{-4}$	$<5 \times 10^{-5}$	1.8×10^{-2}	0.34	$<2 \times 10^{-4}$	$<8 \times 10^{-5}$	$<2 \times 10^{-4}$
D. CONDENSER AND CONDENSATE RETURN	2.46×10^6	$<1 \times 10^{-4}$	7.5×10^{-2}	1.8×10^{-3}	$<2 \times 10^{-5}$	$<7 \times 10^{-4}$	3.2×10^{-3}	$<7 \times 10^{-4}$	$<5 \times 10^{-4}$	$<7 \times 10^{-5}$	2.7×10^{-2}	0.52	$<5.0 \times 10^{-4}$	$<1 \times 10^{-4}$	$<2 \times 10^{-4}$
E. FUEL STORAGE AREA	2.41×10^7	$<7 \times 10^{-5}$	2.1×10^{-2}	4.1×10^{-4}	$<2 \times 10^{-5}$	7.2×10^{-4}	$<2 \times 10^{-4}$	$<4 \times 10^{-4}$	$<7 \times 10^{-5}$	$<2 \times 10^{-5}$	5.8×10^{-4}	9.2×10^{-3}	2×10^{-4}	$<7 \times 10^{-5}$	$<7 \times 10^{-5}$
F. RADWASTE (LIQUID)	1.58×10^7	$<7 \times 10^{-4}$	4.0	7.3×10^{-3}	$<2 \times 10^{-4}$	$<6 \times 10^{-3}$	$<6 \times 10^{-3}$	$<4 \times 10^{-3}$	$<4 \times 10^{-3}$	8.4×10^{-5}	0.21	3.61	$<3 \times 10^{-3}$	$<7 \times 10^{-4}$	$<7 \times 10^{-4}$
TOTAL (Ci)		0.024	88.2	0.046	0.020	0.0067	0.0053	<3.08	<0.02	0.00006	0.362	4.33	0.0052	0.037	<0.01

TABLE C.4.10. (continued) Radionuclide Inventory for Indian Point Station, Unit One, May 1982

	TOTAL ACTIVITY IN CURIES														
	^{155}Eu	^{166m}Ho	^{228}Ac	^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I	^{237}Np	^{238}Pu	^{239,240}Pu	^{241}Am	^{244}Cm	TOTAL
A PRIMARY COOLANT	0 006	0 005	0.06	101	0 17	20	$6 0 \times 10^{3}$	8.9×10^{4}	$2 7 \times 10^{4}$	6×10^{7}	0.0026	0 0046	0.0064	$7 7 \times 10^{4}$	177 8
B SEAL WATER AND PRIMARY MAKEUP	4×10^{5}	5×10^{6}	5×10^{5}	0 01	2×10^{5}	3×10^{3}	9×10^{7}	1×10^{7}	$3 7 \times 10^{8}$	7×10^{4}	4×10^{7}	7×10^{7}	9×10^{7}	1×10^{7}	0 15
C MAIN STEAM LINE	1×10^{4}	3×10^{5}	3×10^{4}	0.09	2×10^{4}	$1 8 \times 10^{2}$	5×10^{6}	8×10^{7}	$2 4 \times 10^{7}$	5×10^{10}	2×10^{6}	5×10^{6}	5×10^{6}	6×10^{7}	0 52
D CONDENSER AND CONDENSATE RETURN	2×10^{4}	5×10^{5}	5×10^{4}	0 14	2×10^{4}	$2 7 \times 10^{2}$	8×10^{6}	1×10^{6}	$3 7 \times 10^{7}$	7×10^{10}	4×10^{6}	7×10^{6}	7×10^{6}	1×10^{6}	0 79
E FUEL STORAGE AREA	5×10^{5}	2×10^{5}	2×10^{4}	0.04	6×10^{5}	7×10^{3}	2×10^{6}	3×10^{7}	$5 9 \times 10^{8}$	2×10^{10}	1×10^{6}	2×10^{6}	2×10^{6}	3×10^{7}	0 079
F RADWASTE (LIQUID)	7×10^{4}	2×10^{4}	3×10^{3}	1 3	$6 8 \times 10^{3}$	1 7	$5 4 \times 10^{4}$	$1 3 \times 10^{5}$	$4 3 \times 10^{6}$	$1 0 \times 10^{7}$	$4 5 \times 10^{5}$	$8 5 \times 10^{5}$	$1 0 \times 10^{4}$	1×10^{5}	10 6
TOTAL (Ci)	0 006	0 005	0 06	102 6	0 177	21 8	0 0065	0 00091	0 00027	1×10^{7}	0 0029	0 0049	0 0065	0 00078	190

TABLE C.4.11. Total Residual Radionuclide Inventory at Indian Point Station, Unit One[a]

		Inventory (Curies)	
Nuclide	Half-life (yr)	Sample Collection (May 25, 1982)	Shutdown (Oct. 31, 1974)
^{55}Fe	2.7	103	720
^{60}Co	5.27	60	160
^{63}Ni	100	22	23
^{137}Cs	30.2	4.4	5.2
^{134}Cs	2.06	0.38	4.8
^{59}Ni	7.5×10^4	0.18	0.18
^{54}Mn	0.854	0.094	44
^{65}Zn	0.668	0.045	115
^{152}Eu	13	0.037	0.055
^{94}Nb	2.0×10^4	0.009	0.009
^{90}Sr	28.8	0.0065	0.0078
^{241}Am	433	0.0065	0.0066
^{108m}Ag	127	0.0053	0.0055
$^{239-240}$Pu	2.41×10^4	0.0049	0.0049
^{238}Pu	87.7	0.0029	0.0031
^{99}Tc	2.14×10^5	0.00091	0.0009
^{244}Cm	18.1	0.00078	0.0010
^{129}I	1.6×10^7	0.00027	0.0003
	Total (Curies)	190	1,070

[a] Excluding neutron-activated pressure vessels and internals, biological shield, concrete surfaces, residues, sludges, and resins in tanks and sumps, and spent fuel.

C.5 RESIDUAL RADIONUCLIDE CONCENTRATIONS AND INVENTORIES AT MONTICELLO

TABLE C.5.1. Residual Radionuclide Concentrations on Hardware from Monticello Nuclear Generating Plant, November 19, 1981

Concentration - pCi/cm^2 (pCi/gm)

Sample Number	Identification	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{94}Nb	^{95}Nb	^{95}Zr	^{103}Ru	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
M-101	Reactor Water Cleanup Piping	79800 (9900)	<17000 (<2200)	690000 (85700)	5780000 (718000)	<9000 (<1100)	<180000 (<22000)	<41000 (<5000)	104000 (12900)	27700 (3400)	<7800 (<900)	<12000 (<1600)	5300 (660)
M-102	Steam Drain Piping	<24 (<1.9)	<14 (<1.1)	52.1 (4.27)	<10 (<0.9)	<2 (<0.2)	<130 (<11)	<1400 (<120)	<80 (<6)	<30 (<2)	<2 (<0.2)	<12 (<1)	<7 (<0.6)
M-103	Condensate Piping	<0.30 (<0.03)	<1.0 (<0.1)	26.7 (3.6)	102 (13.7)	<0.3 (<0.04)	<8 (<1)	<1 (<0.1)	<5 (<0.7)	<2 (<0.2)	<0.7 (<0.1)	<0.8 (<0.1)	27 (3.7)
M-104-L	Condensate Sampling Line to Instrumentation	<0.6 (<0.4)	<2 (<2)	19 (13)	20.8 (14.6)	<0.4 (<0.3)	<30 (<20)	<5 (<4)	<12 (<9)	<4 (<3)	<0.8 (<0.5)	<2 (<1)	<1 (<1)
M-104-V	Valve from Condensate Sampling Line	<0.6 (<0.05)	<2 (<0.2)	18.1 (1.73)	5.3 (0.51)	<0.3 (<0.03)	<20 (<2)	<5 (<0.5)	<12 (<1)	<8 (<0.8)	<0.5 (<0.05)	<2 (<0.2)	<1 (<0.1)
M-105	Turbine Diaphram Corrosion Film Scrapings	<17 (<70)	<70 (<300)	5960 (26000)	27400 (119000)	<60 (<300)	<500 (<2300)	<140 (<590)	<200 (<900)	<100 (<400)	<40 (<180)	<54 (<230)	<30 (<130)
M-106	Sandblasting Grit Used for Cleaning Turbine	-- (<2)	-- (<10)	-- (264)	-- (1280)	-- (<40)	-- (<72)	-- (<19)	-- (<32)	-- (<14)	-- (<10)	-- (<7)	-- (<3)
M-107A	Valve from Condensate Storage Water Line	19.8 --	<10 --	238 --	82 7 --	<2 --	<190 --	<20 --	<66 --	<18 --	<9 --	<9 --	<7 --
M-107B	Valve from Condensate Storage Water Line	4.5 --	<4 --	49.9 --	24.3 --	<0.7 --	<30 --	<8 --	<28 --	<8 --	<4 --	<3 --	<3 --
M-107C	Piping from Condensate Storage Water Line	4.6 (1.2)	<2 (<0.4)	49.5 (13.3)	20.6 (5.53)	<0.4 (<0.1)	<12 (<3)	<3 (<0.9)	<7 (<2)	<2 (<0.6)	<0.8 (<0.2)	<1 (<0.3)	<0.7 (<0.2)
M-108	Filter Screen from Condensate Demin Filter	7.5 (45)	3.3 (20)	181 (1093)	79.1 (477.8)	<1 (<7)	<7 (<40)	<2 (<10)	<3 (<16)	<1 (<7)	<0.9 (<5)	17.3 (104)	<0.2 (<1)
M-109	Pump from Radwaste Drain from Off-Gas Drains	<0.8 --	<4 --	102 --	43.8 --	<1 --	<33 --	<8 --	<90 --	<20 --	<70 --	<3 --	<9 --

TABLE C.5.1. (continued) Residual Radionuclide Concentrations on Hardware from Monticello Nuclear Generating Plant, November 19, 1981

Concentration - pCi/cm^2 (pCi/gm)

Sample Number	Identification	^{126}Sn	^{134}Cs	^{137}Cs	^{141}Ce	^{144}Ce	^{154}Eu	^{155}Eu	^{226}Ra	^{228}Ra
M-101	Reactor Water Cleanup Piping	<7800 (<900)	26900 (3340)	172000 (21400)	<100000 (<13000)	<16000 (2000)	<5100 (<630)	3300 (410)	<3300 (<400)	<16000 (<2000)
M-102	Steam Drain Piping	<2 (<0.2)	5.1 (0.42)	170 (14)	<290 (<24)	<18 (<1)	<0.8 (<0.05)	<6 (<0.5)	<14 (<1)	<16 (<1)
M-103	Condensate Piping	<0.7 (<0.1)	5.9 (0.79)	77.1 (10.4)	310 (42)	<2 (<0.2)	0.046 (0.006)	<0.9 (<0.1)	<0.4 (<0.06)	<1 (<0.1)
M-104-L	Condensate Sampling Line to Instrumentation	<0.8 (<0.5)	<5 (<3)	52.5 (36.7)	<33 (<23)	4.4 (3.1)	<0.2 (<0.1)	<0.8 (<0.6)	<2 (<2)	<3 (<2)
M-104-V	Valve from Condensate Sampling Line	<0.5 (<0.05)	<3 (<0.3)	32.5 (3.11)	<30 (<3)	<3 (<0.3)	<0.1 (<0.01)	<0.8 (<0.08)	<2 (<0.2)	<3 (<0.3)
M-105	Turbine Diaphram Corrosion Film Scrapings	<40 (<180)	<11 (<49)	<7 (<32)	5120 (22341)	<33 (<140)	<36 (<160)	<10 (<44)	<11 (<47)	<69 (<300)
M-106	Sandblasting Grit Used for Cleaning Turbine	-- (<10)	-- (<2)	-- (21.3)	-- (<270)	-- (<6)	-- (<1.5)	-- (<2)	-- (<4)	-- (<10)
M-107A	Valve from Condensate Storage Water Line	<9 --	125 --	1377 --	<140 --	<11 --	<1 --	<3 --	<3 --	<11 --
M-107B	Valve from Condensate Storage Water Line	<4 --	63.7 --	634 --	<60 --	<5 --	<0.4 --	<2 --	<3 --	<4 --
M-107C	Piping from Condensate Storage Water Line	<0.8 (<0.2)	8.5 (2.3)	97.5 (26.1)	<16 (<4)	2 (0.54)	<0.3 (<0.08)	<0.4 (<0.1)	<0.9 (<0.2)	<2 (<0.4)
M-108	Filter Screen from Condensate Demin Filter	<0.9 (<5)	<0.1 (<0.8)	2.22 (13.4)	2970 (17900)	<0.5 (<3)	<0.9 (<5)	<0.1 (<0.9)	<0.2 (<2)	<1 (<6)
M-109	Pump from Radwaste Drain from Off-Gas Drains	<70 --	135 --	12500 --	<180 --	17.9 --	<0.6 --	<5 --	<2 --	<4 --

TABLE C.5.2. Residual Radionuclide Concentrations in Piping and Hardware from Monticello Nuclear Generating Plant, November 1981

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{55}Fe	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I
M-101	Reactor Water Cleanup Line	38,900 ± 390 (4,830 ± 48)	3150 ± 110 (390 ± 13)	1400 ± 64 (170 ± 8)	23 ± 10 (2.9 ± 1.3)	<1 (<0.1)
M-102	Steam Drain Piping	1,500 ± 87 (123 ± 7)	<18 (<2)	<39 (<3)	<81 (<6)	<10 (<0.8)
M-103	Condensate Piping	1,620 ± 18 (220 ± 2)	5.1 ± 1.4 (0.69 ± 0.19)	<1 (<0.1)	<6 (<0.8)	<0.4 (<0.06)
M-104L	Instrument Line Condensate Piping	3,030 ± 150 (2,020 ± 110)	4.6 ± 0.3 (3.2 ± 0.9)	<5 (4)	<53 (<37)	<3 (<2)
M-104V	Instrument Line Condensate Valve	<36 (<3)	<2 (<0.2)	<5 (<0.5)	<26 (<3)	<0.9 (<0.08)
M-105	Turbine Diaphram Scrapings	223 ± 22 (970 ± 97)	5.7 ± 1.9 25 ± 8	<3 (<11)	<12 (<50)	<0.7 (<3)
M-106	Sandblasting Grit from Turbine Cleaning	-- (1,960 ± 240)	-- (<8)	-- (<8)	-- (<40)	-- (<0.9)
M-107A	Condensate Storage Water Line - Valve	166 ± 50 --	5.0 ± 1.9 --	<10 --	<15 --	<0.4 --
M-107B	Condensate Storage Water Line - Valve	42 ± 15 --	<6 --	<10 --	<46 --	<1 --
M-107C	Condensate Storage Water Line - Piping	52 ± 9 (14 ± 3)	9.0 ± 3.0 (2.4 ± 0.8)	<2 (<0.5)	<7 (<2)	<0.4 (<0.1)
M-108	Screen from Condensate Demin. Filter	1.7 ± 1.2 10 ± 7	<0.2 (<1)	0.47 ± 0.11 (2.8 ± 0.6)	<2 (<10)	<0.01 (<0.08)
M-109	Pump from Rad-Waste Drain (Off-Gas)	89 ± 26 --	<1 --	312 ± 9 --	<9 --	<0.4 --

TABLE C.5.3. Residual Radionuclide Concentrations in Piping and Hardware from Monticello Nuclear Generating Plant, November 1981

Radionuclide Concentration - pCi/cm^2 (pCi/gm) in parentheses

Sample	Identification	^{238}Pu	$^{239-240}$Pu	^{241}Am	^{244}Cm	^{242}Cm
M-101	Reactor Water Cleanup Line	87 ± 2 (11 ± 0.2)	87 ± 2 (11 ± 0.2)	216 ± 6 (28.8 ± 0.8)	182 ± 6 (22.6 ± 0.7)	86.7 ± 10 (10.8 ± 1.2)
M-102	Steam Drain Piping	<0.09 (<0.007)	<0.07 (<0.006)	1.8 ± 0.6 (0.15 ± 0.05)	<0.7 (<0.05)	<3 (<0.3)
M-103	Condensate Piping	0.049 ± 0.010 (0.0066 ± 0.0013)	0.012 ± 0.005 (0.0016 ± 0.0007)	0.35 ± 0.05 (0.047 ± 0.007)	0.28 ± 0.04 (0.038 ± 0.006)	<0.2 (<0.02)
M-104L	Instrument Line Condensate Piping	<0.02 (<0.02)	<0.01 (<0.01)	0.41 ± 0.12 (0.28 ± 0.08)	<0.06 (<0.04)	<0.6 (<0.4)
M-104V	Instrument Line Condensate Valve	<0.03 (<0.003)	<0.02 (<0.002)	<0.3 (<0.03)	<0.1 (<0.01)	<0.8 (<0.08)
M-105	Turbine Diaphram Scrapings	0.42 ± 0.04 (1.8 ± 0.2)	0.46 ± 0.04 (2.0 ± 0.2)	0.74 ± 0.10 (3.2 ± 0.4)	0.25 ± 0.06 (1.1 ± 0.3)	<0.4 (<1.7)
M-106	Sandblasting Grit from Turbine Cleaning	-- (<0.06)	-- (<0.04)	-- (0.39 ± 0.14)	-- (<0.1)	-- (<1)
M-107A	Condensate Storage Water Line - Valve	<0.01 --	<0.01 --	0.24 ± 0.05 --	<0.06 --	<0.3 --
M-107B	Condensate Storage Water Line - Valve	<0.009 --	<0.006 --	<0.5 --	<0.2 --	<1.6 --
M-107C	Condensate Storage Water Line - Piping	0.0086 ± 0.0025 (0.0023 ± 0.0007)	0.0049 ± 0.0019 (0.0013 ± 0.0005)	<0.1 (<0.03)	<0.06 (<0.02)	<0.6 (<0.2)
M-108	Screen from Condensate Demin. Filter	0.0035 ± 0.0004 (0.021 ± 0.002)	0.0028 ± 0.0003 (0.017 ± 0.002)	0.011 ± 0.0004 (0.006 ± 0.0021)	<0.003 (<0.02)	<0.03 (<0.2)
M-109	Pump from Radwaste Drain (Off-Gas)	0.21 ± 0.01 --	0.15 ± 0.01 --	0.18 ± 0.04 --	<0.1 --	<0.5 --

TABLE C.5.4. Residual Radionuclide Concentrations on Hardware from Monticello Nuclear Generating Plant, May 4, 1981

Concentration - pCi/cm^2 (pCi/gm)

Sample Number	Identification	^{51}Cr	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{94}Nb	^{103}Ru	^{106}Ru	^{108m}Ag	^{110m}Ag
M-100-1	Feedwater Heater #14A impinger plate	108 (55.8)	3.76 (1.94)	2.09 (1.08)	744 (385)	5859 (3030)	< 0.9 (< 0.5)	7.79 (4.03)	71.9 (37.2)	< 0.2 (< 0.1)	< 2 < 1
M-100-2	Steel tubing from #14A Feedwater heater impinger plate	297 (926)	5.86 (7.30)	7.03 (8.76)	566 (705)	1537 (1914)	< 0.2 (< 0.3)	3.98 (4.96)	7.87 (9.80)	< 0.2 (< 0.3)	3.2 (4.0)
M-100-3	Extension nuts from Condens Demin Element Assembly	1.75 (0.626)	0.90 (0.25)	< 0.04 (< 0.01)	36.0 (12.9)	7.39 (2.65)	< 0.03 (< 0.01)	< 0.05 (< 0.02)	0.56 (0.20)	< 0.04 (< 0.02)	< 0.2 (< 0.06)
M-100-4	Guide Rods from Condens Demin Element Assembly	< 0.2 (< 0.2)	0.23 (0.23)	< 0.02 < 0.02	4.86 (4.90)	0.403 (0.405)	< 0.01 (< 0.01)	< 0.03 (< 0.03)	< 0.2 (< 0.2)	< 0.05 (< 0.02)	< 0.06 (< 0.06)
M-100-5	Springs from Condens Demin Element Assembly	10.0 (11.9)	0.14 (0.16)	< 0.04 (< 0.04)	13.3 (15.7)	10.0 (11.8)	< 0.03 (< 0.04)	< 0.06 (< 0.07)	< 0.4 (< 0.5)	< 0.04 (< 0.05)	< 0.2 (< 0.2)
M-100-6	Spring Seats from Condens Demin Element Assembly	2.7 (5.3)	1.55 (3.02)	0.12 (0.23)	42.1 (81.7)	7.74 (15.0)	< 0.04 (< 0.08)	< 0.07 (< 0.1)	1.5 (2.9)	< 0.05 (< 0.1)	< 0.2 (< 0.4)
M-100-7*	Silver plated gasket seal for Reactor Vessel Head	53.0 (29.8)	344 (193)	598 (337)	2908 (1637)	429 (241)	< 0.4 (< 0.2)	7.27 (4.09)	< 4.0 (< 2.0)	< 0.3 (< 0.2)	< 2 < 1

Sample Number	^{126}Sn	^{125}Sb	^{134}Cs	^{137}Cs	^{141}Ce	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{241}Am
M-100-1	< 0.3 (< 0.1)	24.5 (12.7)	< 0.7 (< 0.2)	2.06 (1.06)	1428 (739)	10.2 (5.26)	< 1 (< 0.5)	< 3.0 (< 3.0)	1.74 (0.902)	2.91 (1.51)
M-100-2	< 0.3 (< 0.4)	17.8 (22.2)	1.7 (2.1)	0.68 (0.85)	692 (862)	0.91 (1.14)	< 1 (< 1)	< 4.0 (< 5.0)	1.1 (1.3)	0.25 (0.31)
M-100-3	< 0.05 (< 0.02)	1.7 0.62	8.95 (3.21)	76.7 (27.5)	15.6 (5.61)	0.28 (0.10)	< 1 (< 0.04)	< 0.5 (< 0.2)	0.078 (0.028)	0.27 (0.098)
M-100-4	< 0.02 (< 0.02)	0.97 (0.98)	4.2 (4.2)	37.0 (37.3)	< 0.03 (< 0.03)	0.20 (0.21)	< 0.06 (< 0.06)	< 0.2 (< 0.2)	< 0.04 (< 0.04)	< 0.05 (< 0.05)
M-100-5	< 0.04 (< 0.06)	4.8 (5.6)	2.2 (2.6)	20.5 (24.2)	53.3 (62.9)	< 0.3 (< 0.4)	< 0.1 (< 0.2)	< 0.5 (< 0.6)	< 0.2 (< 0.2)	0.14 (0.16)
M-100-6	< 0.06 (< 0.1)	1.2 (2.4)	8.3 (16.3)	75.6 (147)	13.6 (26.4)	< 0.2 (< 0.5)	< 0.2 (< 0.3)	< 0.7 (< 1)	< 0.1 (< 0.2)	0.021 (0.040)
M-100-7	< 0.3 (< 0.2)	< 0.9 (< 0.5)	< 0.4 (< 0.2)	5.26 (2.96)	3.48 (1.96)	3.93 (2.21)	< 1 (< 1)	< 6 (< 3)	< 0.6 (< 0.4)	2.2 (1.3)

*Sample M-100-7 contained 255 pCi/cm^2 and (143 pCi/gm) of ^{131}I.

TABLE C.5.5. Residual Radionuclide Concentrations on Hardware from Monticello Nuclear Generating Plant, May 4, 1981

Concentration - pCi/cm^2 (pCi/gm)

Sample Number	Identification	^{55}Fe	^{63}Ni	^{238}Pu	$^{239-240}$Pu	^{241}Am	^{242}Cm	^{244}Cm
M-100-1	Feedwater Heater #14A Impinger Plate	1450 (750)	13 (6.7)	2.70 (1.39)	2.23 (1.15)	2.59 (1.34)	0.031 (0.016)	0.65 (0.34)
M-100-2	Steel Tubing from #14A Feedwater Heater Impinger Plate	4200 (5240)	18 (22)	0.56 (0.70)	0.52 (0.65)	0.32 (0.40)	0.0085 (0.011)	0.089 (0.11)
M-100-3	Extension Nuts from Condens Demin Element Assembly	294 (105)	<1 (<0.5)	0.46 (0.16)	0.10 (0.038)	0.12 (0.043)	0.0030 (0.0011)	0.11 (0.040)
M-100-4	Guide Rods from Condens Demin Element Assembly	122 (123)	<0.1 (<0.1)	0.017 (0.017)	0.0065 (0.0065)	0.012 (0.012)	0.0010 (0.0010)	0.015 (0.015)
M-100-5	Springs from Condens Demin Element Assembly	51 (60)	<0.6 (<0.7)	0.13 (0.15)	0.061 (0.072)	0.069 (0.081)	0.0042 (0.0050)	0.070 (0.083)
M-100-6	Springs Seats from Condens Demin Element	727 (1410)	<1 (<2)	0.069 (0.13)	0.037 (0.073)	0.036 (0.070)	0.0017 (0.0033)	0.026 (0.051)
M-100-7	Silver Plated Gasket Seal for Reactor Vessel	17600 (9880)	67.6 (38.1)	0.16 (0.092)	0.11 (0.061)	0.15 (0.086)	0.13 (0.073)	0.16 (0.092)

TABLE C.5.6. Residual Radionuclide Concentrations in Monticello Nuclear Generating Plant Concrete Cores

Concentration in pCi/cm^2*

Sample Number	Identification	Core Depth (cm)	Sample Weight	^{54}Mn	^{60}Co	^{65}Zn	^{95}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
MCC-1	Reactor Bldg, 1001'2" Elev. Tool Decon area	0-1	170.0	12.8 ± 0.8	922 ± 9	18 ± 3	<1	<10	<7.6	<3.4	<2.1
		1-2	175.0	0.078 ± 0.054	0.28 ± 0.11	<0.19	<0.043	<0.45	<0.14	0.29 ± 0.20	<0.43
MCC-2	Turbine Bldg, Operating Floor, Near Feedwater Heater	0-1	170.0	<0.056	0.48 ± 0.12	<0.19	<0.041	<0.43	<0.48	<0.22	<0.12
		1-2	185.0	<0.056	0.35 ± 0.09	<0.19	<0.050	<0.51	<0.44	<0.20	0.26 ± 0.12
MCC-3	Reactor Bldg, 1001'2" Elev. Tool Decon Area	0-1	220.0	2.1 ± 0.5	388 ± 2	3.6 ± 1.9	<0.31	<3.4	<3.1	<2.3	<0.88
		1-2	175.0	<0.063	1.9 ± 0.2	<0.23	0.064 ± 0.043	0.63 ± 0.46	1.1 ± 0.4	<0.25	<0.43
MCC-4	Radwaste Bldg, 935'0" Elev. Shipping area	0-1	155.0	2.5 ± 0.5	198 ± 1	51 ± 4	0.39 ± 0.33	<6.8	<11	<1.9	<3.2
		1-2	170.0	<0.056	1.6 ± 0.2	<0.22	<0.042	<0.51	<0.45	<0.30	<0.13
MCC-5	Radwaste Bldg. 935'0" Elev Compactor Area	0-1	165.0	1.0 ± 0.2	40.7 ± 0.6	2.6 ± 0.7	<0.12	<2.2	<3.6	<0.81	<1.0
		1-2	175.0	<0.061	1.1 ± 0.6	<0.22	<0.044	<0.50	<0.44	<0.21	<0.12
MCC-6	Turbine Bldg. Operating Floor, Condensate Demin Work Area	0-1	160.0	0.085 ± 0.061	0.57 ± 0.13	<0.21	0.069 ± 0.040	<0.49	1.2 ± 0.6	<0.21	<0.16
		1-2	170.0	<0.054	<0.080	<0.16	<0.036	0.58 ± 0.43	<0.38	<0.16	<0.40
MCC-7	Reactor Bldg, 1027'8" Elev, Refuel Floor	0-1	180.0	0.65 ± 0.16	30.0 ± 0.6	<0.57	<0.10	<1.1	2.3 ± 1.2	<0.69	<0.33
		1-2	185.0	<0.054	<0.098	<0.20	<0.040	<0.43	<0.52	<0.18	<0.13
MCC-8	Reactor Bldg, 962'6" Elev, Reactor Water Cleanup Pump Room	0-1	185.0	60 ± 3	8790 ± 10	276 ± 10	<3.0	190 ± 20	<15	<13	<9.2
		1-2	185.0	<0.50	184 ± 2	<1.3	<0.22	19 ± 3	<2.2	<2.0	<0.62
MCC-9	Reactor Bldg, 985'6" Elev, Reactor Radwaste Pump Room	0-1	160.0	4.2 ± 0.3	63.8 ± 0.9	12 ± 1	<0.15	<2.5	<3.8	<1.0	<1.1
		1-2	175.0	0.064 ± 0.054	<0.10	<0.18	0.091 ± 0.044	<0.45	<0.39	<0.40	<0.43
MCC-10	Reactor Bldg, 985'6" Elev, Snubber Mainten. Shop	0-1	155.0	<0.060	1.7 ± .1	<0.8	<0.047	<0.52	<0.59	<0.26	<0.17
		1-2	175.0	<0.066	.14 ± .09	<0.2	<0.037	<0.51	<0.47	<0.20	<0.14

*To obtain pCi/gm, multiply (pCi/cm^2) by 71 and then divide by the weight.

TABLE C.5.6. (continued) Residual Radionuclide Concentrations in Monticello Nuclear Generating Plant Concrete Cores

Concentration in pCi/cm^2*

	Identification	Core Depth (cm)	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ac
MCC-1	Reactor Bldg, 1001'2" Elev, Tool Decon Area	0-1	<0.66	192 ± 6	1750 ± 10	6.8 ± 3.6	<1.4	5.0 ± 1.0	<1.4	<1	<5.5
		1-2	<0.047	<0.064	<0.065	<0.40	0.38 ± 0.24	<0.14	<0.16	<0.053	1.4 ± 0.4
MCC-2	Turbine Bldg, Operating Floor, Near Feedwater Heater	0-1	<0.043	<0.063	0.77 ± 0.09	<0.40	<0.24	<0.14	<0.16	<0.052	1.3 ± 0.4
		1-2	<0.041	<0.063	<0.062	<0.41	<0.20	<0.15	<0.17	<0.058	2.1 ± 0.5
MCC-3	Reactor Bldg, 1001'2" Elev, Tool Decon Area	0-1	<0.27	10.4 ± 0.4	122 ± 1	4.7 ± 1.7	<0.94	<1.0	2.1 ± 0.6	<0.43	<3.7
		1-2	<0.040	<0.063	<0.067	<0.39	<0.30	<0.14	<0.15	<0.090	1.1 ± 0.5
MCC-4	Radwaste Bldg, 935'0" Elev, Shipping Area	0-1	<1.0	550 ± 2	1640 ± 10	<5.1	<1.8	<0.56	<2.0	<0.48	<3.1
		1-2	<0.046	0.12 ± 0.06	0.33 ± 0.08	<0.39	<0.20	<0.14	<0.14	<0.060	0.86 ± 0.47
MCC-5	Radwaste Bldg. 935'0" Elev, Compactor Area	0-1	<0.32	39.3 ± 0.6	474 ± 2	<1.7	<0.44	0.39 ± 0.25	<0.64	<0.16	<1.3
		1-2	<0.044	<1.8	<0.067	<0.41	<0.22	<0.13	0.16 ± 0.15	0.13 ± 0.06	1.8 ± 0.5
MCC-6	Turbine Bldg, Operating Floor, Condensate Demin Work Area	0-1	<0.056	0.38 ± 0.08	39.2 ± 0.2	<0.43	<0.30	<0.14	<0.16	<0.054	2.5 ± 0.5
		1-2	<0.036	<0.060	0.16 ± 0.05	<0.38	<0.22	<0.12	<0.14	0.14 ± 0.04	1.1 ± 0.4
MCC-7	Reactor Bldg, 1027'8" Elev, Refuel Floor	0-1	<0.10	2.8 ± 0.2	23.0 ± 0.4	<0.68	<0.40	<0.22	<0.26	<0.14	<1.1
		1-2	<0.044	<0.061	<0.063	<0.43	0.42 ± 0.26	<0.16	<0.16	<0.065	1.6 ± 0.4
MCC-8	Reactor Bldg, 962'6" Elev, Reactor Water Cleanup Pump Room	0-1	<1.3	69.5 ± 2.7	545 ± 3	26 ± 8	9.3 ± 4.7	1517 ± 3.4	10 ± 3	<2.3	<21
		1-2	<0.19	2.9 ± 0.4	24.1 ± 0.5	1.8 ± 1.3	<0.46	<1.0	0.89 ± 0.50	<0.31	<2.6
MCC-9	Reactor Bldg, 985'6" Elev, Reactor Radwaste Pump Room	0-1	<0.50	50.4 ± 0.7	493 ± 2	<1.8	<0.50	0.40 ± 0.30	1.8 ± 0.7	<0.21	<1.6
		1-2	<0.044	<0.060	0.21 ± 0.07	<0.40	<0.22	<0.15	<0.15	<0.055	1.2 ± 0.4
MCC-10	Reactor Bldg, 985'6" Elev, Snubber Mainten. Shop	0-1	<0.055	0.64 ± 0.13	6.4 ± 0.2	<0.45	<0.25	<0.12	<0.16	<0.069	1.4 ± 0.5
		1-2	<0.044	<1.8	<0.064	<0.43	<0.23	<0.14	<0.16	<0.056	2.5 ± 0.5

*To obtain pCi/gm, multiply (pCi/cm^2) by 71 and then divide by the weight.

TABLE C.5.7. Radionuclide Concentrations in Surficial Soils at Monticello Nuclear Generating Plant, May 6, 1981

Concentration (pCi/gm)

Sample Number	Location	^{7}Be	^{40}K	^{51}Cr	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{95}Zr	^{95}Nb	^{103}Ru	^{106}Ru
M-SS-1	20m NW of Plant	0.33	11.2	<0.05	<0.004	<0.004	0.083	<0.01	0.14	0.28	0.047	<0.07
M-SS-2	20m W of Plant	0.25	11.6	<0.1	<0.004	<0.005	<0.006	<0.01	0.11	0.24	0.034	<0.05
M-SS-3	20m S of Plant	0.42	12.8	<0.07	<0.024	<0.006	0.446	<0.02	0.19	0.42	0.042	<0.07
M-SS-4	20m E of Plant	0.25	12.7	<0.06	<0.005	<0.006	0.094	<0.01	0.15	0.31	0.054	<0.06

Sample Number	Location	^{125}Sb	^{134}Cs	^{137}Cs	^{141}Ce	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{228}Th	^{226}Ra	^{228}Ra
M-SS-1	20m NW of Plant	<0.02	<0.005	0.29	0.03	0.11	<0.04	<0.03	<0.02	0.45	0.48	0.52
M-SS-2	20m W of Plant	<0.02	<0.004	0.068	0.03	0.083	<0.04	<0.03	<0.03	0.32	0.25	0.39
M-SS-3	20m S of Plant	<0.03	0.16	2.05	0.04	0.17	<0.06	<0.04	<0.02	0.37	0.30	0.46
M-SS-4	20m E of Plant	<0.03	<0.005	0.25	0.03	0.13	<0.05	<0.03	<0.02	0.33	0.27	0.40

TABLE C.5.8. Residual Radionuclide Concentrations Associated with Major Piping and Hardware Used for Inventory Estimates

Concentration - μCi/cm^2

Piping System	^{54}Mn	^{55}Fe	^{60}Co	^{63}Ni	^{65}Zn	^{90}Sr	^{103}Ru	^{106}Ru
Reactor Water Piping$^{(a)}$	0.080	0.039	0.69	0.0032	5.78	0.0014	0.10	0.028
Steam Piping$^{(b)}$	$<1.7 \times 10^{-5}$	2.3×10^{-4}	0.0060	5.7×10^{-6}	0.027	$<3 \times 10^{-6}$	$<2 \times 10^{-4}$	$<1 \times 10^{-4}$
Condensate Piping$^{(c)}$	2.5×10^{-6}	8.4×10^{-4}	3.1×10^{-4}	7.0×10^{-6}	6.2×10^{-5}	$<1.5 \times 10^{-6}$	$<6 \times 10^{-6}$	$<2 \times 10^{-6}$
Feedwater Heaters$^{(d)}$	4.9×10^{-6}	2.8×10^{-3}	6.6×10^{-4}	1.6×10^{-5}	3.7×10^{-3}	$<1 \times 10^{-8}$ (est.)	5.9×10^{-6}	4.0×10^{-5}

(a) Derived from analyses of reactor water cleanup line (Sample M-101).

(b) Derived from analyses of turbine diaphram scrapings (Sample M-105).

(c) Derived from averaging the analyses of condensate piping (Sample M-103), and condensate storage water line (M-107B).

(d) Derived from averaging the analyses of feedwater heater impinger plate (Sample M-100-1) and piping from the impinger plate (Sample M-100-2).

TABLE C.5.8. (continued) Residual Radionuclide Concentrations Associated with Major Piping and Hardware Used for Inventory Estimates

Concentration - $\mu Ci/cm^2$

	^{125}Sb	^{134}Cs	^{137}Cs	^{155}Eu	^{238}Pu	$^{239-240}Pu$	^{241}Am	^{244}Cm
Reactor Water Piping(a)	0.0053	0.027	0.17	0.0033	8.7×10^{-5}	8.7×10^{-5}	2.2×10^{-4}	1.8×10^{-4}
Steam Piping(b)	$< 3 \times 10^{-5}$	5.9×10^{-6}	7.7×10^{-5}	$< 1 \times 10^{-6}$	4.2×10^{-7}	4.6×10^{-7}	7.4×10^{-7}	2.5×10^{-7}
Condensate Piping(c)	1.4×10^{-5}	7.2×10^{-5}	8.8×10^{-5}	$< 1 \times 10^{-6}$	2.9×10^{-8}	8.5×10^{-9}	6.8×10^{-7}	1.7×10^{-7}
Feedwater Heaters(d)	2.1×10^{-5}	1.2×10^{-6}	1.3×10^{-6}	^{141}Ce 1.1×10^{-3} ^{155}Eu 1.4×10^{-6}	1.6×10^{-6}	1.4×10^{-6}	1.5×10^{-6}	2.0×10^{-8}

(a) Derived from analyses of reactor water cleanup line (Sample M-101).

(b) Derived from analyses of turbine diaphram scrapings (Sample M-105).

(c) Derived from averaging the analyses of condensate piping (Sample M-103), and condensate storage water line (M-107B).

(d) Derived from averaging the analyses of feedwater heater impinger plate (Sample M-100-1) and piping from the impinger plate (Sample M-100-2).

TABLE C.5.9. Residual Radionuclide Inventories in Various Piping Systems at Monticello - 1981

PIPING	TOTAL SURFACE AREA (cm^2)(A)	CURIES ^{54}Mn	^{55}Fe	^{60}Co	^{63}Ni	^{65}Zn	^{90}Sr	^{103}Ru	^{106}Ru	^{125}Sb	^{134}Cs	^{137}Cs	^{155}Eu	^{238}Pu	^{239,240}Pu	^{241}Am	^{244}Cm
PIPING CATEGORY																	
STEAM/AIR	1.36×10^8	<0.0023	0.031	0.82	7.8×10^{-4}	3.70	$<4.1 \times 10^{-4}$	<0.027	<0.014	<0.0041	8.0×10^{-4}	0.010	$<1.4 \times 10^{-4}$	5.7×10^{-5}	6.3×10^{-5}	1.0×10^{-4}	3.4×10^{-5}
REACTOR WATER	1.10×10^7	0.89	0.43	7.60	0.035	84	0.015	1.10	0.31	0.058	0.30	1.90	0.038	9.8×10^{-4}	9.8×10^{-4}	0.0024	0.0020
CONDENSATE	1.92×10^8	4.8×10^{-4}	0.18	0.059	0.0013	0.012	$<2.9 \times 10^{-4}$	<0.0012	$<3.8 \times 10^{-4}$	0.0037	0.014	0.017	$<1.9 \times 10^{-4}$	5.8×10^{-5}	1.8×10^{-5}	1.3×10^{-4}	3.3×10^{-5}
REACTOR BUILDING EQUIP																	
RWCU REGEN. HEAT EX RWCU NON REGEN. HEAT. EX RHR HEAT EX. REACTOR VESSEL	4.5×10^7 (B)	3.6	1.8	31	0.14	282	0.061	4.5	1.3	0.23	1.2	7.8	0.15	0.0039	0.0038	0.0098	0.0082
FUEL POOL HEAT EX SKIMMER SURGE TANKS FUEL POOL, REACTOR WELL, DRYER & SEPARATOR POOL, RBCC WATER HEAT EX	4.1×10^7 (C)	2×10^{-4}	0.11	0.027	6.6×10^{-4}	0.15	$<4.1 \times 10^{-7}$	2.4×10^{-4}	0.0016	8.6×10^{-4}	4.9×10^{-5}	5.3×10^{-5}	5.7×10^{-5}	6.6×10^{-5}	5.7×10^{-5}	6.2×10^{-5}	8.2×10^{-7}
TURBINE GENERATOR BUILDING																	
MAIN CONDENSER, SJAE CONDENSER, GSS CONDENSER, CONDENSATE STORAGE TANKS, EVAP. DRAIN TANKS, MOISTURE SEP. DRAIN TANK SEAL WATER LIQUID TANK, PUMPED DRAIN TANK	8.26×10^8 (D)	0.0021	0.69	0.25	0.0056	0.052	$<1.2 \times 10^{-3}$	<0.006	$<1.6 \times 10^{-3}$	0.012	0.060	0.073	$<8.2 \times 10^{-4}$	2.4×10^{-5}	6.9×10^{-6}	5.6×10^{-4}	1.4×10^{-4}
LP & HP FEEDWATER HEATERS REHEATER DRAIN TANKS, MOISTURE SEPARATOR REHEATERS	1.11×10^9 (C)	0.0054	3.0	0.73	0.018	4.1	$<1.1 \times 10^{-5}$	0.0065	0.043	0.023	0.0013	0.0014	0.0015	0.0018	0.0016	0.0017	2.2×10^{-5}
MAIN TURBINE, STEAM EVAP., TURBINE BYPASS VALVE ASSEMB	4.62×10^7 (E)	$<7.8 \times 10^{-4}$	0.011	0.28	2.8×10^{-4}	1.3	$<1.4 \times 10^{-4}$	<0.0092	<0.0048	<0.0014	2.7×10^{-4}	0.0034	$<4.8 \times 10^{-5}$	1.9×10^{-5}	2.1×10^{-5}	3.4×10^{-5}	1.2×10^{-5}
RADWASTE AND CONTROL BUILDING																	
WASTE COLLECTOR TANK, WASTE SAMPLE TANK, FLOOR DRAIN COLLECTOR TANK, FLOOR DRAIN SAMPLE TANK, CHEMICAL WASTE TANKS, DISTILLATE TANKS, DETERGENT DRAIN TANK	7.8×10^6 (D)	2.0×10^{-5}	0.0085	0.0024	5.3×10^{-5}	4.9×10^{-4}	$<1.2 \times 10^{-5}$	$<4.9 \times 10^{-5}$	1.5×10^{-5}	1.1×10^{-4}	5.7×10^{-4}	$<6.9 \times 10^{-4}$	$<7.7 \times 10^{-6}$	2.3×10^{-7}	6.5×10^{-8}	5.3×10^{-6}	1.3×10^{-6}
CONDENSATE PHASE SEP. TANKS, CONDENSATE BACKWASH RECEIVER TANK, WASTE SURGE TANK, WASTE SLUDGE PHASE SEP TANK, DECON SOL'N CONC WASTE TANK, SPENT RESIN TANK, CLEANUP PHASE SEP. TANKS, DECON SOLUTION CONCENTRATOR	6.39×10^6 (B)	0.51	0.25	4.4	0.020	37	0.0087	0.64	0.18	0.34	0.17	1.1	0.021	5.6×10^{-4}	5.6×10^{-4}	0.0014	0.0012
TOTAL FOR PLANT		5.0	6.5	45	0.22	370	0.085	6.2	1.8	0.67	1.7	11	0.21	0.0072	0.0069	0.016	0.012

(A) OBTAINED BY SUMMING INTERNAL PIPING SURFACE AREAS GIVEN IN TABLE E.2-5 IN Oak, et al. (1980)

(B) USED RADIONUCLIDE CONCENTRATIONS MEASURED ON REACTOR WATER CLEANUP PIPING (M-101)

(C) USED RADIONUCLIDE CONCENTRATIONS MEASURED ON FEEDWATER HEATER HARDWARE (M-100-1 & M-100-2 AVG.)

(D) USED RADIONUCLIDE CONCENTRATIONS MEASURED ON CONDENSATE PIPING (M-103) AND CONDENSATE STORAGE WATER LINE (M-107-B)

(E) USED RADIONUCLIDE CONCENTRATIONS MEASURED ON TURBINE DIAPHRAM SCRAPINGS (M-105)

TABLE C.5.10. Total Residual Radionuclide Inventory at Monticello Nuclear Generating Plant-1981[a]

Radionuclide[b]	Half-Life (years)	Inventory (curies)
^{65}Zn	0.668	370
^{60}Co	5.27	45
^{137}Cs	30.2	11
^{55}Fe	2.7	6.5
^{103}Ru	0.108	6.2
^{54}Mn	0.855	5.0
^{106}Ru	1.01	1.8
^{134}Cs	2.06	1.7
^{125}Sb	2.77	0.67
^{63}Ni	100	0.22
^{155}Eu	4.96	0.21
^{90}Sr	28.5	0.085
^{241}Am	432	0.016
^{244}Cm	18	0.012
^{238}Pu	87.7	0.0072
$^{239-240}Pu$	24,110	0.0069
	Total	448

(a) Excluding neutron activated pressure vessel and internals, biological shield, concrete surfaces, residues, sludges and resins in tanks and sumps, and spent fuel.

(b) Other long-lived radionuclides, specifically listed in 10 CFR 61, e.g., ^{94}Nb, ^{99}Tc and ^{129}I, were not included in the inventory because of their insignificant concentrations in the residual radioactive corrosion films in the plant piping and equipment.

C.6 RESIDUAL RADIONUCLIDE CONCENTRATIONS AND INVENTORIES AT TURKEY POINT

TABLE C.6.1. Residual Radionuclide Concentrations in Turkey Point Reactor Hardware Corrosion Films, October, 1981

Concentrations in pCi/cm^2
( ) in pCi/g

Sample	^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb
TP-H1 Steam Dryer	2.6 ± 0.7 (2.0 ± 0.5)	188 ± 2 (142 ± 2)	10 ± 2 (8 ± 2)	<0.1 (<0.08)	<3 (<3)	<1 (<0.8)	<3 (<3)	241 ± 2 (183 ± 2)
TP-H2 Steam Dryer	<0.9 (<0.1)	121 ± 1 (18.3 ± 0.2)	4 ± 2 (0.7 ± 0.4)	<0.1 (<0.2)	<5 (<0.7)	<1 (<0.2)	12 ± 5 (1.9 ± 0.7)	5 ± 1 (0.8 ± 0.2)
TP-H3 Main Steam Line	2 ± 1	193 ± 1	7 ± 2	<0.1	<7	<2	<7	2 ± 1
TP-H4 Secondary Feedwater Line	<0.7	42 ± 1	3 ± 2	<0.1	<3	<1	<7	<1
TP-H5 Control Rod Ventilation Duct	<0.3	4.3 ± 0.3	<1	<0.07	<2	<1	<2	<1
TP-H6 Steam Generator Blowdown Line Secondary	<0.2 (<0.06)	37.5 ± 0.3 (10.9 ± 0.1)	1.9 ± 0.6 (0.6 ± 0.2)	<0.03 (<0.009)	<1 (<0.3)	<0.4 (<0.1)	<1 (<0.3)	0.9 ± 0.4 (0.3 ± 0.1)
TP-H7 Alumina grit	(20200 ± 700)	(1.15 X 10^6 ± 0.003 X 10^6)	(49,000 ± 2,000)	(<100)	(13,000 ± 3,000)	(<1,000)	(<4,000)	(4,000 ± 1,000)
TP-H8 "C" Steam Generator	100 ± 40	2520 ± 10	90 ± 10	<1	<20	<10	<20	<10
TP-H9 "C" Steam Generator	40 ± 5	1860 ± 10	110 ± 10	<1	<20	<10	<30	<10

TABLE C.6.1. (continued) Residual Radionuclide Concentrations in Turkey Point Reactor Hardware Corrosion Films, October, 1981

Concentrations in pCi/cm^2
( ) in pCi/g

Sample	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ac
TP-H1 Steam Dryer	<0.09 (<0.07)	<0.3 (<0.3)	<0.2 (<0.1)	<1 (<0.9)	<0.3 (<0.3)	<0.5 (<0.4)	<0.2 (<0.1)	<0.1 (<0.1)	<2 (<1)
TP-H2 Steam Dryer	<0.1 (<0.02)	<0.7 (<0.1)	1.3 ± 0.3 (0.20 ± 0.05)	<2 (<0.4)	<0.5 (<0.07)	<0.5 (<0.07)	<0.5 (<0.07)	<0.2 (<0.3)	<2 (<0.4)
TP-H3 Main Steam Line	<0.1	<0.7	1.5 ± 0.3	<2	<0.7	<0.7	<0.3	<0.2	<2
TP-H4 Secondary Feed-water Line	<0.1	<0.7	1.2 ± 0.2	<2	<0.7	<0.7	<0.3	<0.2	<2
TP-H5 Control Rod Ventilation Duct	<0.1	0.59 ± 0.03	16.3 ± 0.3	3 ± 2	<0.7	<0.7	<0.3	<0.1	<1
TP-H6 Steam Generator Blowdown Line Secondary	<0.04 (<0.01)	<0.2 (<0.06)	1.32 ± 0.08 (0.38 ± 0.02)	<0.4 (<0.2)	0.5 ± 0.2 (0.13 ± 0.04)	0.4 ± 0.2 (0.11 ± 0.05)	<0.1 (<0.03)	<0.05 (<0.02)	<0.6 (<0.2)
TP-H7 Alumina Grit	(<100)	(<400)	(<200)	(9,000 ± 1,000)	(600 ± 400)	(<500)	(<300)	(<200)	(<2,000)
TP-H8 "C" Steam Generator	<0.5	5 ± 2	<1	<10	<2	<2	<1	<1	<10
TP-H9 "C" Steam Generator	<1	<3	<2	<10	<2	<3	<2	<1	<10

TABLE C.6.2. Residual Radionuclide Concentrations in Turkey Point Reactor Hardware Samples, October, 1981

Concentrations in pCi/cm^2
( ) in pCi/g

	^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I
TP-3 Main Steam Line	63 ± 8	<0.2	1.9 ± 0.3	<0.01	<0.09	<0.04
TP-4 Secondary Feedwater Line	1,500 ± 200	0.8 ± 0.1	33 ± 3	<0.008	0.08 ± 0.01	<0.09
TP-5 Control Rod Ventilation Duct	6 ± 1	0.05 ± 0.02	0.24 ± 0.02	0.025 ± 0.004	0.05 ± 0.01	<0.08
TP-7 Alumina Grit	(1.49 X 10^6 ± 0.09 X 10^6)	(180 ± 10)	(5,100 ± 400)	(42 ± 3)	(310 ± 20)	(<0.9)
TP-8 "C" Steam Generator	430 ± 30	0.81 ± 0.08	2.1 ± 0.2	0.39 ± 0.03	0.50 ± 0.03	<0.1
TP-9 "C" Steam Generator	1110 ± 80	<4	200 ± 20	0.076 ± 0.008	0.34 ± 0.07	<0.09

TABLE C.6.3. Residual Transuranic Radionuclide Concentrations in Turkey Point Reactor Hardware Samples, October, 1981

Concentrations in pCi/cm^2
( ) in pCi/g

	^{237}Np	^{238}Pu	^{239,240}Pu	^{241}Am	^{242}Cm	^{244}Cm
TP-3 Main Steam Line	<0.0008	0.0048 ± 0.0006	0.0031 ± 0.0004	0.0074 ± 0.0008	<0.009	0.0017 ± 0.0005
TP-4 Secondary Feed-Water Line	0.0028 ± 0.00009	0.0021 ± 0.0004	0.0027 ± 0.0004	0.0073 ± 0.0015	<0.003	0.0013 ± 0.0009
TP-5 Control Rod Ventilation Duct	0.0017 ± 0.0004	0.0006 ± 0.0003	0.0014 ± 0.0003	0.0019 ± 0.0006	<0.002	0.0005 ± 0.0003
TP-7 Alumina Grit	(0.09 ± 0.01)	(90 ± 4)	(63 ± 3)	(86 ± 4)	(28 ± 5)	(51 ± 3)
TP-8 "C" Steam Generator	0.0013 ± 0.001	0.056 ± 0.003	0.043 ± 0.003	0.15 ± 0.02	0.03 ± 0.01	0.091 ± 0.007
TP-9 "C" Steam Generator	0.0014 ± 0.0005	0.058 ± 0.003	0.047 ± 0.003	0.14 ± 0.009	0.09 ± 0.04	0.074 ± 0.006

TABLE C.6.4. Radionuclide Concentrations in Top Two Centimeters of Turkey Point Concrete Cores

(Concentration in pCi/cm^2)

	Depth (cm)	Sample Weight (g)	^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ac
TP CC-1	0-1	143	<0.2	38.3 ± 0.8	<1	<0.1	<2	<3	<1	<1	<0.3	34.8 ± 0.7	161 ± 1	<2	<0.5	<0.3	<0.6	<0.2	<2
	1-2	165	<0.08	1.2 ± 0.2	<0.3	<0.07	<1	<2	<0.3	<0.4	<0.1	8.0 ± 0.3	38.1 ± 0.6	<0.8	<0.3	<0.2	<0.3	<.08	<0.5
TP CC-2	0-1	130	0.094 ± 0.038	29.5 ± 0.1	<0.1	<0.03	<0.4	<0.7	<0.2	<0.2	<0.06	44.0 ± 0.1	197 ± 1	0.71 ± 0.30	<0.1	<0.05	<0.1	<0.04	<0.3
	1-2	154	<0.02	0.53 ± 0.03	<0.06	<0.01	<0.2	<0.3	<0.06	<0.09	<0.03	4.80 ± 0.06	23.4 ± 0.1	<0.2	0.31 ± 0.07	<0.04	<0.07	<0.02	0.51 ± 0.12
TP CC-3	0-1	165	188 ± 3	5150 ± 10	<9	<2	<23	54 ± 27	29 ± 11	37 ± 8	<2	440 ± 8	7320 ± 10	<12	<9	4	<5	<2.5	<19
	1-2	158	<0.01	0.44 ± 0.03	<0.05	<0.01	<0.1	<0.2	<0.05	<0.04	<0.01	1.33 ± 0.03	4.86 ± 0.05	<0.1	0.25 ± 0.06	<0.03	<0.04	<0.01	0.44 ± 0.08
TP CC-4	0-1	147	2.4 ± 1.0	819 ± 4	6.9 ± 3.6	<1	<11	<15	<4	<5	<1	970 ± 4	3750 ± 10	<7	<3	<1	<3	<1	<7
	1-2	155	<0.2	8.1 ± 0.4	<0.5	<0.1	<3	<4	<0.6	<1	<0.4	81 ± 1	312 ± 2	<2	<0.5	<0.2	<0.8	<0.2	<1
TP CC-5	0-1	135	98 ± 2	3330 ± 10	<7	<1	<14	<13	<9	12 ± 4	<1	294 ± 3	906 ± 4	<6	<3	<2	<3	<2	<14
	1-2	162	<0.03	3.47 ± 0.07	<0.1	<0.03	<0.5	1.9 ± 0.8	<0.1	<0.2	<0.07	52.2 ± 0.2	163 ± 1	<0.4	<0.2	<0.05	<0.2	<0.04	<0.2
TP CC-6	0-1	188	<0.05	14.6 ± 0.2	<0.2	<0.04	0.98 ± 0.42	<0.5	<0.2	0.54 ± 0.14	<0.4	2.81 ± 0.09	18.5 ± 0.2	<0.3	<0.4	<0.08	<0.1	<0.05	<0.4
	1-2	120	<0.01	0.56 ± 0.03	<0.06	<0.01	<0.1	<0.2	<0.05	<0.05	<0.02	<0.5	4.36 ± 0.05	<0.1	0.17 ± 0.06	<0.03	<0.04	<0.01	<0.1
TP CC-7	0-1	140	89 ± 2	4910 ± 10	<8	<1	<20	<20	<10	102 ± 11	<2	692 ± 4	2940 ± 10	<9	<4	<3	<4	<2	<20
	1-2	140	<0.06	53.5 ± 0.2	<0.2	<0.04	<0.6	<0.9	<0.3	<0.2	0.22 ± 0.08	42.6 ± 0.2	210 ± 1	<0.4	0.25 ± 0.10	<0.08	<0.2	<0.06	<0.4
TP CC-8	0-1	159	4.1 ± 0.5	155 ± 2	<2	<0.4	<6	<8	8 ± 2	20 ± 5	<0.8	364 ± 2	1170 ± 10	<6	<1	<0.6	<2	<0.5	<4
	1-2	151	<0.05	0.37 ± 0.08	<0.2	<0.04	<0.4	<0.5	<0.2	<0.2	<0.05	1.2 ± 0.1	6.6 ± 0.2	<0.4	0.5 ± 0.2	<0.1	<0.1	<0.05	<0.3
TP CC-9	0-1	159	11 ± 3	572 ± 5	<8	<2	<40	<60	75 ± 11	<20	<5	7910 ± 20	24100 ± 100	<25	<15	<4	<10	<3	<20
	1-2	161	<0.2	1.7 ± 0.4	<0.7	<0.2	<5	<8	<0.9	<2	<0.7	150 ± 2	453 ± 3	<4	2.2 ± 0.7	<0.3	<1.4	<0.3	4 ± 1
TP CC-10	0-1	193	138 ± 3	4120 ± 10	<8	<2	<20	<30	24 ± 10	280 ± 10	<3	2820 ± 10	11300 ± 100	<13	<7	<3	<5	<2	<20
	1-2	173	9.1 ± 0.9	880 ± 3	<3	<0.5	<6	<7	<4	148 ± 5	<0.6	99.5 ± 1.2	410 ± 2	<3	<1	<1	<1	<0.8	<6
	2-3	152	<0.03	12.8 ± 0.1	<0.1	<0.02	<0.2	<0.3	<0.1	<0.08	<0.03	5.00 ± 0.06	22.5 ± 0.1	<0.2	0.28 ± 0.06	<0.04	<0.06	<0.03	0.5 ± 0.2
TP CC-11	0-1	142	13 ± 3	8000 ± 10	<10	<2	<20	<20	<10	<5	<1	112 ± 2	394 ± 3	<9	<5	<4	<3	<3	<20
	1-2	148	<0.3	145 ± 1	<1	<0.2	<2	<2	<1	<0.5	<0.2	9.9 ± 0.4	35.1 ± 0.5	<1	<0.4	<0.4	<0.4	<0.3	<2
TP CC-12	0-1	164	316 ± 7	17200 ± 100	<20	<4	<40	<40	<30	232 ± 25	<3	959 ± 6	1910 ± 10	<20	<10	<9	<8	15 ± 5	<50
	1-2	171	0.21 ± 0.06	74.9 ± 0.2	<0.2	<0.04	<0.5	<0.5	<0.3	<0.1	<0.05	13.4 ± 0.1	70.7 ± 0.2	<0.3	<0.1	<0.0[illegible]	<0.1	<0.05	<0.4
TP CC-13	0-1	138	5.4 ± 1.0	1700 ± 10	<4	<0.6	<7	<6	11 ± 5	20 ± 4	<0.5	12.4 ± 0.7	58 ± 1	<3	<2	<1	<1	<0.9	<8
	1-2	171	<0.1	44.8 ± 0.4	<0.4	<0.07	<0.7	<0.6	<0.4	<0.2	<0.06	0.66 ± 0.10	2.9 ± 0.1	<0.4	<0.2	<0.1	<0.2	<0.09	<0.7
TP CC-14	0-1	165	1.7 ± 0.1	191 ± 1	<0.3	<0.07	<0.8	<1	<0.4	1.1 ± [illegible]	<0.9	55.1 ± 0.2	309 ± 1	<0.5	<0.2	<0.1	<0.2	<0.09	<0.7
	1-2	127	<0.02	4.92 ± 0.06	<0.06	<0.01	<0.1	<0.2	<0.07	[illegible]	<0.02	1.12 ± 0.03	8.35 ± 0.06	<0.1	0.25 ± 0.05	<0.03	<0.04	<0.02	9.47 ± 0.13

To obtain pCi/g, multiply by 71 and divide by sample weight.

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TABLE C.6.5. Radionuclide Concentration in Turkey Point Cores Collected from Beneath the Reactor Vessel

(Concentration in pCi/cm^2)

	Depth (cm)	Sample Weight (g)	^{54}Mn	^{60}Co	^{65}Zn	^{94}Nb	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho	^{228}Ac
TP CC-15	0-1	170	180 ± 10	15600 ± 100	1720 ± 40	<4	<50	<40	<30	68 ± 13	<4	1240 ± 10	5180 ± 10	90 ± 30	3120 ± 30	457 ± 12	<11	<6	<30
	1-2	194	<4	2600 ± 10	260 ± 10	<3	<30	<30	<20	<8	<3	468 ± 5	1260 ± 10	160 ± 20	4840 ± 30	620 ± 10	<8	<4	<20
	2-3	165	<4	2360 ± 10	240 ± 10	<3	<30	<30	<15	<7	<3	291 ± 5	604 ± 6	170 ± 20	5060 ± 30	613 ± 9	25 ± 8	11 ± 4	<20
	3-4	167	<4	2170 ± 10	240 ± 10	<3	<30	<25	<15	<7	<3	217 ± 4	368 ± 5	160 ± 10	4940 ± 30	574 ± 9	22 ± 8	10 ± 4	<20
	4-5	161	<4	2180 ± 10	240 ± 10	<3	<30	<20	<15	<7	<2	206 ± 4	282 ± 5	195 ± 20	4620 ± 30	504 ± 8	24 ± 8	9 ± 4	<20
	5-6	173	<4	2360 ± 10	250 ± 10	<3	<30	<25	<15	<7	<3	180 ± 4	182 ± 5	140 ± 20	5030 ± 30	546 ± 9	<8	12 ± 4	<20
	6-7	213	<5	3180 ± 10	390 ± 20	<3	<40	<30	<20	<9	<3	235 ± 5	165 ± 5	220 ± 20	7040 ± 30	[illegible]	<10	10 ± 4	<20
TP CC-16	0-1	159	120 ± 7	12400 ± 100	4070 ± 20	<4	<50	<40	<30	90 ± 30	<4	12100 ± 100	2580 ± 10	130 ± 20	3480 ± 30	524 ± 11	<10	<6	<20
	1-2	185	<4	2420 ± 10	250 ± 10	<3	<30	<30	<20	<8	<3	268 ± 5	129 ± 5	120 ± 20	5410 ± 30	705 ± 10	<9	<4	<20
	2-3	151	<4	2350 ± 10	280 ± 10	<3	<30	<25	<15	<7	<2	184 ± 4	64 ± 4	150 ± 20	5050 ± 30	607 ± 9	<8	<4	<20
	3-4	181	<4	2420 ± 10	280 ± 10	<3	<30	<30	<20	<8	<3	196 ± 5	51 ± 5	180 ± 20	5900 ± 30	700 ± 10	<9	<4	<20
	4-5	127	<5	5030 ± 20	240 ± 10	<3	<30	<30	<20	<8	<3	120 ± 4	22 ± 5	140 ± 20	4100 ± 30	470 ± 10	<8	<4	<20
	5-6	139	30 ± 6	9540 ± 20	300 ± 15	<4	<40	<30	<25	<9	<3	135 ± 5	16 ± 6	140 ± 20	4480 ± 30	500 ± 10	<9	[illegible]	<20
	6-7	194	21 ± 5	6630 ± 20	320 ± 20	<3	<40	<30	<20	<9	<3	153 ± 5	22 ± 5	180 ± 20	5160 ± 30	560 ± 10	<9	<5	<20
	7-8	170	<4	2630 ± 10	305 ± 20	<3	<30	<30	<20	<9	<3	158 ± 4	14 ± 5	210 ± 20	6250 ± 30	660 ± 10	<9	<4	<20

To obtain pCi/g, multiply by 71 and divide by sample weight.

TABLE C.6.6. Turkey Point Unit 4 Concrete Core (TP-CC-15) Taken Directly Beneath Vessel Activity at Shutdown and Projected to 30 EFPY (pCi/gm)

	Concentration (pCi/gm)						
	0-1 cm	1-2 cm	2-3 cm	3-4 cm	4-5 cm	5-6 cm	6-7 cm
^{46}Sc	1001	1350	1390	1380	1360	1450	1300
^{51}Cr	8140	4830	1835	626	580	440	450
^{54}Mn	76	3.1	3.1	4.0	1.8	<1.8	<1.8
^{58}Co	165	19	22	19	17	23	25
^{59}Fe	670	780	820	780	780	810	720
^{60}Co	10100	1700	1400	1420	1430	1540	1320
^{63}Ni							<16
^{65}Zn	630	61	53	54	59	52	50
^{94}Nb	<3	<2	<2	<2	<2	<2	<2
^{108m}Ag	<3	<2	<2	<2	<2	<2	<2
^{110m}Ag	13	<5	<5	<5	<5	<5	<5
^{124}Sb	49	20	19	17	12	8.5	12.5
^{134}Cs	500	188	116	85	80	72	58
^{137}Cs	6550	1610	772	471	353	224	133
^{141}Ce	63	98	85	80	80	89	54
^{152}Eu	2490	4010	4170	4090	3860	4210	3670
^{154}Eu	346	470	470	390	370	440	350
^{155}Eu	<13	<42	<30	<27	<27	<33	<33
^{166m}Ho	<6	<20	<14	<13	<13	<16	<16

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TABLE C.6.7. Turkey Point Unit 4 Concrete Core (TP-CC-16) Taken Directly Beneath Vessel (TR-J) Activity at Shutdown and Projected to 30 EFPY (pCi/gm)

	Concentrations (pCi/gm)							
	0-1 cm	1-2 cm	2-3 cm	3-4 cm	4-5 cm	5-6 cm	6-7 cm	7-8 cm
^{46}Sc	1024	1266	1323	1560	1190	2203	1502	1703
^{51}Cr	3667	1390	760	890	580	670	490	850
^{54}Mn	49	<2	2.7	<2	7.6	12.5	8.9	<1.8
^{59}Fe	680	740	780	890	2120	4510	2850	950
^{60}Co	8090	1590	1540	1708	3300	6226	4310	1710
^{65}Zn	2420	79	83	83	80	84	84	100
^{94}Nb	<1.7	<0.6	<0.6	<0.6	<1.1	<0.6	<0.6	<0.6
^{108m}Ag	<0.8	<0.6	<0.6	<0.6	<1.1	<0.6	<0.6	<0.6
^{110m}Ag	<10	<8	<5	<7	<7	<9	<7	<5
^{124}Sb	72	31	21	22	12	17	17	22
^{134}Cs	530	115	77	86	53	57	67	67
^{137}Cs	3190	165	90	76	32	28	35	27
^{141}Ce	54	63	67	38	76	103	98	72
^{152}Eu	3041	4440	4080	4870	3400	3730	4370	5280
^{154}Eu	810	1100	950	1170	740	775	980	1100
^{166m}Ho	<13	<20	<23	<20	<23	<33	<10	<15

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TABLE C.6.8. Residual Radionuclide Concentrations in Turkey Point Concrete, October, 1981 (Concentration in pCi/cm^2)

		^{55}Fe	^{59}Ni	^{63}Ni	^{90}Sr	^{99}Tc	^{129}I
TPCC-9	0-1 cm	1250 ± 90	4 ± 1	39 ± 5	0.24 ± 0.02	1.9 ± 0.1	<0.4
TPCC-15	0-1 cm	28,000 ± 1000	150 ± 10	-	1.6 ± 0.1	2.2 ± 0.3	<0.3
	1-2 cm	36,000 ± 2000	<3	70 ± 7	0.57 ± 0.05	2.4 ± 0.3	0.4 ± 0.3
	2-3 cm	31,000 ± 2000	2.7 ± 0.7	19 ± 2	0.56 ± 0.04	1.0 ± 0.1	<0.5
	3-4 cm	34,000 ± 2000	<3	-	1.0 ± 0.1	0.72 ± 0.08	0.5 ± 0.3

TABLE C.6.9. Residual Transuranium Radionuclide Concentrations in Turkey Point Concrete, October, 1981

(Concentrations in pCi/cm^2)

		^{237}Np	^{238}Pu	^{239,240}Pu	^{241}Am	^{242}Cm	^{244}Cm
TPCC-9	0-1 cm	0.027 ± 0.01	0.040 ± 0.005	0.012 ± 0.003	0.015 ± 0.006	0.04 ± 0.02	0.023 ± 0.006
TPCC-15	0-1 cm	0.010 ± 0.002	0.025 ± 0.005	0.54 ± 0.03	0.10 ± 0.01	0.06 ± 0.03	0.05 ± 0.01
	1-2 cm	0.010 ± 0.003	0.004 ± 0.003	0.61 ± 0.02	0.029 ± 0.009	<0.03	0.011 ± 0.005
	2-3 cm	0.008 ± 0.002	0.002 ± 0.001	0.42 ± 0.02	0.013 ± 0.006	<0.04	0.014 ± 0.005
	3-4	0.016 ± 0.004	<0.002	0.41 ± 0.02	0.011 ± 0.007	<0.04	0.232 ± 0.009

TABLE C.6.10. Radionuclide Concentrations in Turkey Point Nuclear Plant Soils, October, 1981

(Concentrations in pCi/g dry weight)

		^{40}K	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{103}Ru	^{106}Ru	^{125}Sb	^{134}Cs	^{137}Cs	^{141}Ce	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu
TP-1	0-4 cm	2.8 ± 0.1	0.12 ± 0.02	<0.04	35.7 ± 0.1	<0.07	<0.05	<0.2	<0.09	0.28 ± 0.02	1.68 ± 0.02	<0.07	<0.07	<0.1	<0.2	<0.04
TP-1	4-8 cm	2.5 ± 0.1	0.02 ± 0.01	<0.02	4.66 ± 0.03	<0.03	<0.02	<0.07	<0.04	<0.008	0.31 ± 0.01	0.06 ± 0.04	<0.04	<0.06	<0.08	<0.02
TP-2	0-4 cm	2.8 ± 0.1	0.22 ± 0.02	0.32 ± 0.04	28.0 ± 0.1	<0.06	<0.05	<0.1	<0.08	0.98 ± 0.02	3.63 ± 0.03	<0.07	0.27 ± 0.06	<0.1	<0.1	<0.03
TP-2	4-8 cm	2.2 ± 0.1	<0.008	0.03 ± 0.01	2.48 ± 0.03	<0.02	<0.02	<0.06	<0.04	<0.008	0.34 ± 0.01	<0.04	<0.03	<0.06	<0.05	<0.02
TP-3	0-4 cm	2.9 ± 0.1	0.03 ± 0.01	<0.02	4.07 ± 0.04	<0.03	<0.04	<0.1	<0.08	1.07 ± 0.02	6.48 ± 0.04	0.10 ± 0.06	<0.05	<0.1	<0.07	<0.03
TP-3	4-8 cm	2.8 ± 0.2	0.02 ± 0.01	<0.02	1.24 ± 0.03	<0.04	<0.05	<0.1	<0.07	0.37 ± 0.02	2.87 ± 0.04	<0.07	<0.07	<0.1	<0.07	<0.06
TP-4	0-4 cm	1.67 ± 0.09	<0.005	<0.01	0.089 ± 0.006	0.33 ± 0.01	<0.03	<0.04	<0.04	<0.007	0.132 ± 0.005	0.18 ± 0.07	0.08 ± 0.03	<0.04	<0.03	<0.01
TP-4	4-8 cm	2.3 ± 0.1	<0.05	<0.01	0.016 ± 0.005	<0.01	<0.03	<0.04	<0.03	<0.08	0.086 ± 0.005	<0.07	<0.03	<0.04	<0.02	<0.01
TP-5	0-4 cm	2.6 ± 0.2	<0.01	<0.02	0.16 ± 0.02	<0.03	0.07 ± 0.04	<0.08	<0.07	<0.01	0.19 ± 0.01	<0.07	<0.05	<0.09	<0.06	<0.03
TP-5	4-8 cm	2.4 ± 0.1	<0.004	<0.008	0.019 ± 0.004	<0.01	<0.02	<0.04	<0.03	<0.005	0.092 ± 0.006	0.07 ± 0.03	<0.03	<0.04	<0.02	<0.01
TP-6	0-4 cm	2.7 ± 0.1	<0.006	<0.01	0.167 ± 0.009	<0.02	<0.04	<0.05	<0.04	<0.007	0.187 ± 0.007	<0.08	<0.03	<0.05	<0.03	<0.02
TP-6	4-8 cm	2.8 ± 0.1	<0.004	<0.008	0.020 ± 0.004	<0.01	<0.02	<0.03	<0.03	<0.005	0.100 ± 0.005	0.07 ± 0.03	<0.02	<0.04	<0.02	<0.01
TP-7	0-4 cm	2.8 ± 0.1	<0.005	<0.009	0.021 ± 0.006	<0.02	<0.02	<0.05	<0.04	<0.007	0.088 ± 0.006	<0.04	<0.05	<0.05	<0.03	<0.02
TP-7	4-8 cm	2.6 ± 0.1	<0.004	<0.008	0.021 ± 0.005	<0.01	<0.02	<0.04	<0.03	<0.006	0.097 ± 0.005	0.07 ± 0.03	<0.03	<0.04	<0.03	<0.01
TP-8	0-4 cm	3.2 ± 0.1	0.16 ± 0.02	0.07 ± 0.03	14.0 ± 0.1	<0.04	<0.05	0.2 ± 0.1	<0.06	1.82 ± 0.02	4.12 ± 0.02	<0.06	0.24 ± 0.05	<0.1	<0.09	<0.03
TP-8	4-8 cm	2.8 ± 0.1	0.34 ± 0.03	<0.05	45.4 ± 0.1	<0.08	<0.09	<0.2	19.1 ± 0.2	5.22 ± 0.03	11.25 ± 0.04	0.2 ± 0.1	<0.09	<0.2	<0.2	<0.06
TP-9	0-5 cm	2.2 ± 0.1	0.08 ± 0.02	0.10 ± 0.05	27.7 ± 0.1	<0.06	<0.1	0.2 ± 0.1	11.2 ± 0.1	2.74 ± 0.02	6.01 ± 0.03	<0.1	<0.07	<0.1	<0.1	<0.03
TP-10	0-5 cm	2.7 ± 0.1	0.11 ± 0.03	<0.07	40.7 ± 0.1	<0.08	<0.1	<0.2	22.3 ± 0.2	5.60 ± 0.03	11.41 ± 0.05	0.3 ± 0.2	0.2 ± 0.09	<0.2	0.3 ± 0.2	<0.05
TP-SEDIMENT		4.5 ± 0.2	0.03 ± 0.01	<0.03	1.82 ± 0.03	<0.03	<0.06	0.18 ± 0.08	<0.06	<0.02	0.18 ± 0.01	0.2 ± 0.1	<0.08	<0.08	<0.08	<0.02

TABLE C.6.11. Radionuclide Compositions Used for Inventory Calculations

Radionuclide	Composition A	Composition B
Mn-54	0.25 uci/cm^2	2 pCi/cm^2
Fe-55	18.74	780
Co-60	14.46	120
Ni-59	0.002	1
Ni-63	0.064	20
Zn-65	0.62	5
Nb-94	<0.001	<0.1
Sr-90	0.0005	<0.01
Tc-99	0.004	<0.1
Ru-106	0.16	<5
Ag-108m	<0.01	<2
Ag-110m	<0.05	<7
Sb-125	0.05	60
Sn-126	<0.001	<0.09
I -129	<1 x 10 e-5	<0.09
Cs-134	<0.005	<0.7
Cs-137	<0.003	1.3
Ce-144	0.11	<2
Eu-152	0.008	<0.7
Eu-154	<0.006	<0.7
Eu-155	<0.004	<0.3
Ho-166m	<0.003	<0.2
Ac-228	<0.03	<2
Np-237	1.1 x 10 e-6	<0.0008
Pu-238	0.0011	0.0035
Pu-239,240	0.0008	0.0029
Am-241	0.0011	0.0074
Cm-242	0.0004	<0.009
Cm-244	0.0006	0.0015

TABLE C.6.12. Estimated Surface Areas for Turkey Point Reactor Systems

System	Surface Area (m^2)
Steam generators	1.4 x 10 e4
Pressurizer	8.7 x 10 e1
Piping (except RCS)	1.1 x 10 e3
RCS Piping	1.4 x 10 e2
Secondary	2.3 x 10 e5
Radwaste	1.4 x 10 e3

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TABLE C.6.13. Total Estimated Inventory for Turkey Point Unit 3* by Radionuclide-October, 1981[(a)(b)]

Concentrations in Curies

Radionuclide	Total Inventory
Mn-54	10
Fe-55	790
Co-58	1110
Co-60	610
Ni-59	0.1
Ni-63	3
Zn-65	26
Sr-90	0.02
Tc-99	0.2
Ru-106	7
Sb-125	2
Cs-134	<0.2
Cs-137	<0.1
Ce-144	5
Eu-152	0.3
Pu-238	0.05
Pu-239,240	0.03
Am-241	0.05
Cm-242	0.01
Cm-244	0.03

Total Estimated Inventory - 2580 Curies

* Unit 4 inventory can be estimated by multiplying figures in the table by 0.7.

(a) Excluding neutron activated pressure vessel and internals; biological shield; concrete surfaces; residues, sludges and resins in tanks and sumps; and spent fuel.

(b) Other long-lived radionuclides specifically listed in 10 CFR 61 (e.g., ^{94}Nb - ^{129}I) were not included in the inventory because of their insignificant concentrations in the residual radioactive corrosion films in the plant piping and equipment.

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C.7 RESIDUAL RADIONUCLIDE CONCENTRATIONS AND INVENTORIES AT RANCHO SECO

TABLE C.7.1. Radionuclide Concentrations (pCi/cm^2 and pCi/g) Associated with Rancho Seco Reactor System Hardware

	HIGH PRESSURE INJECTION FOR PRIMARY COOLANT RSH-1 (3/83)		HEPA FILTER FROM PRIMARY SIDE OF STEAM GENERATOR REPAIR TENT RSH-2 (6/81)	PRIMARY SYSTEM LETDOWN COOLER PIPING RSH-4A (6/83)		PRIMARY SYSTEM LETDOWN COOLER VALVE RSH-4V (6/83)	SECONDARY SYSTEM CHEVRON BAFFLE PLATE MAIN STEAM REHEATERS RSH-3 (3/83)	
192 d	pCi/cm^2	pCi/g	pCi/cm^2	pCi/cm^2	pCi/g	pCi/cm^2	pCi/cm^2	pCi/g
^{54}Mn	140 ± 10	15 ± 1	204,000 ± 2,000	7,100 ± 100	790 ± 10	28,700 ± 300	26.6 ± 0.9	62 ± 2
^{55}Fe	70,800 ± 700	7,800 ± 80	980,000 ± 70,000	100,000 ± 1,000	11,100 ± 100	147,000 ± 10,000	710 ± 50	1,600 ± 100
^{57}Co	27 ± 3	2.9 ± 0.3	23,000 ± 400	510 ± 10	68 ± 1	2,350 ± 40	0.98 ± 0.09	2.3 ± 0.2
^{58}Co	720 ± 60	80 ± 7	1.18 × 10^8 ± 1.42 × 10^6	158,100 ± 700	17,580 ± 80	48,100 ± 600	108 ± 4	252 ± 9
^{60}Co	5,560 ± 60	610 ± 10	225,000 ± 1,000	22,700 ± 200	2,510 ± 20	132,100 ± 700	64 ± 1	149 ± 3
^{59}Ni	75 ± 1	8.3 ± 0.1	430 ± 7	513 ± 4	58.2 ± 0.5	581 ± 7	0.63 ± 0.04	1.47 ± 0.09
^{63}Ni	13,110 ± 70	1,443 ± 8	770,000 ± 2,000	88,600 ± 200	9,770 ± 20	76,200 ± 300	118 ± 2	274 ± 4
^{65}Zn	26 ± 19	3 ± 2	2,400 ± 1,600	450 ± 50	50 ± 6	300 ± 100	<0.4	<0.9
^{90}Sr	<10	<1	1,710 ± 80	<40	<4	<20	<0.009	<0.02
^{95}Zr	<60	<6	N.D.	<100	<10	<200	<1	<3
^{94}Nb	<0.5	<0.06	<40	<20	<2	<60	<0.2	<0.4
^{95}Nb	<700	<20	N.D.	400 ± 200	50 ± 20	<400	<5	<10
^{99}Tc	<9	<1	<5	<20	<2	<2	<0.4	<1
^{106}Ru	<200	<20	16,000 ± 8,000	<800	<90	3,000 ± 2,000	<4	<10
^{108m}Ag	<6	<0.7	<30	<20	<2	<40	<0.3	<0.6
^{110m}Ag	[illegible]	190 ± 20	185,000 ± 10,000	18,600 ± 400	2,060 ± 60	21,000 ± 1,000	<2	<5
^{125}Sb	[illegible]	44 ± 3	14,900 ± 400	9,000 ± 100	1,000 ± 10	6,800 ± 200	9.3 ± 0.6	22 ± 1
^{129}I	[illegible]	<0.002	<0.03	<0.03	<0.004	<0.06	<0.0008	<0.002
^{134}Cs	<20	<2	800 ± 400	<50	<6	800 ± 200	<0.3	<0.8
^{137}Cs	<8	<0.9	3,900 ± 100	<30	<3	2,740 ± 90	0.4 ± 0.1	0.8 ± 0.3
^{141}Ce	400 ± 200	50 ± 20	N.D.	<100	<10	<200	<4	<10
^{144}Ce	<20	<2	6,000 ± 1,000	<70	<8	300 ± 200	<0.6	<1
^{238}Pu	2.38 ± 0.03	0.250 ± 0.003	20.7 ± 0.4	1.60 ± 0.03	0.164 ± 0.004	1.62 ± 0.04	0.0023 ± 0.0002	0.0053 ± 0.0004
$^{239\text{-}240}$Pu	1.48 ± 0.02	0.162 ± 0.003	12.4 ± 0.3	0.82 ± 0.02	0.089 ± 0.002	0.84 ± 0.03	0.0012 ± 0.0001	0.0028 ± 0.0003
^{241}Am	1.56 ± 0.02	0.172 ± 0.003	9.8 ± 0.6	1.87 ± 0.08	0.20 ± 0.01	1.4 ± 0.1	0.0009 ± 0.0002	0.0020 ± 0.0005
^{242}Cm	0.31 ± 0.02	0.034 ± 0.002	700 ± 30	7.2 ± 0.2	0.79 ± 0.02	9.4 ± 0.4	0.0013 ± 0.0003	0.0030 ± 0.0006
^{244}Cm	1.04 ± 0.02	0.115 ± 0.002	8.6 ± 0.5	1.81 ± 0.08	0.200 ± 0.009	1.7 ± 0.1	0.0015 ± 0.0002	0.0[illegible]4 ± 0.0004

N.D. = NON-DETECTABLE

TABLE C.7.2. Residual Radionuclide Concentrations (pCi/cm^2 and pCi/g) Associated with Rancho Seco Reactor Secondary Scarping Samples

	RS-SC-1 Steam Generator OSG-B Secondary Side (3-83)	RS-SC-2 Steam Generator OSG-A Secondary Side (3-83)	RS-SC-3 Boron Residue Turbine Bldg. Wall (3-83)		RS-SC-4 Boron Residue Underneath Fuel Transfer Chute (3-83)		RS-SC-5 Inside of Turbine Housing (3-83)	
	pCi/cm^2	pCi/cm^2	pCi/cm^2	pCi/g	pCi/cm^2	pCi/g	pCi/cm^2	pCi/g
^{54}Mn	<0.2	22.3 ± 0.4	<0.02	<4	81 ± 3	3,380 ± 100	0.08 ± 0.01	23 ± 3
^{58}Co	4 ± 2	80 ± 3	<0.2	<60	4,880 ± 90	180,000 ± 2,000	0.4 ± 0.1	100 ± 40
^{60}Co	51.6 ± 0.5	881 ± 1	0.05 ± 0.01	14 ± 3	513 ± 4	20,080 ± 200	0.46 ± 0.01	118 ± 4
^{65}Zn	<6.7	7.3 ± 0.9	<0.05	<10	<8	<200	<0.06	<9
^{95}Zr	<4	<6	<0.5	<100	<90	<3,000	<0.3	<80
^{95}Nb	<30	<2	<3	<2,000	<1,000	<50,000	9 ± 5	2,000 ± 1,000
^{106}Ru	<1	<2	<0.1	<40	<30	<1,000	<0.07	<20
^{108m}Ag	<0.05	<0.09	<0.005	<3	<2	<80	<0.004	<1
^{110m}Ag	<0.9	<1	0.12 ± 0.08	30 ± 20	<10	<400	<0.04	<10
^{124}Sb	<0.2	<0.4	9.8 ± 0.1	2,800 ± 20	153 ± 8	6,100 ± 300	0.26 ± 0.02	142 ± 6
^{125}Sb	<0.07	<0.1	<0.01	<3	<2	<90	<0.004	<1
^{134}Cs	1.4 ± 0.2	10.7 ± 0.2	<0.01	<3	2,701 ± 9	180,200 ± 400	0.012 ± 0.007	3 ± 2
^{137}Cs	2.8 ± 0.1	22.2 ± 0.2	0.032 ± 0.01	9 ± 2	4,884 ± 9	190,400 ± 400	0.010 ± 0.004	3 ± 1
^{141}Ce	<60	120 ± 70	<30	<8,000	–	–	<10	<3,000
^{144}Ce	<0.9	2 ± 1	<0.2	<40	<20	<800	<0.06	<20
^{152}Eu	<0.2	<0.3	0.14 ± 0.01	40 ± 10	<7	<200	<0.02	<6
^{154}Eu	<0.9	<1	<0.08	<20	<10	<900	<0.05	<10
^{155}Eu	<0.4	<0.6	<0.04	<10	<4	<200	<0.01	<4
^{166m}Ho	0.3 ± 0.1	<0.2	<0.01	<3	<2	<70	<0.007	<2

TABLE C.7.3. Radionuclide Concentrations (pCi/cm^2)* in Rancho Seco Concrete - March, 1983

CONCRETE CORE	^{54}Mn	^{58}Co	^{60}Co	^{65}Zn	^{103}Ru	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho
RS-CC-1	35 ± 1	380 ± 10	590 ± 10	17 ± 5	900 ± 500	<20	<2	59 ± 8	<7	<2	1.570 ± 10	11.370 ± 10	40 ± 20	24 ± 3	<9	<3	12 ± 1
RS-CC-2	11.1 ± 0.3	280 ± 3	270 ± 4	38 ± 1	<40	17 ± 3	<0.2	71 ± 1	4.0 ± 0.9	1.7 ± 0.2	280 ± 1	1,720 ± 10	5 ± 2	58.4 ± 0.6	<2	<0.5	16.1 ± 0.3

*TO CONVERT pCi/cm^2 TO pCi/g MULTIPLY BY 0.49

TABLE C.7.4. Radionuclide Concentrations (pCi/g) in Rancho Seco Soils - March, 1983

SOILS	^{22}Na	^{54}Mn	^{58}Co	^{60}Co	^{51}Cr	^{65}Zn	^{103}Ru	^{106}Ru	^{108m}Ag	^{110m}Ag	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho
RS-SS-1 BORATED WATER STORAGE TANK	<0.009	0.27 ± 0.01	<0.07	11.32 ± 0.04	<10	<0.05	<0.3	<0.08	<0.006	<0.06	0.75 ± 0.02	<0.006	0.95 ± 0.02	4.94 ± 0.02	<0.06	<0.02	<0.1	<0.02	<0.01
RS-SS-2 DEMIN. REACTOR COOLANT STORAGE TANK	<0.006	0.008 ± 0.004	<0.02	0.025 ± 0.004	<3	<0.01	<0.08	<0.03	<0.002	<0.01	<0.007	<0.002	0.094 ± 0.004	0.485 ± 0.006	<0.02	<0.01	<0.03	<0.008	<0.004
RS-SS-3 TANK FARM YARD N.E. END	0.008 ± 0.004	<0.003	<0.02	0.004 ± 0.002	<3	<0.01	<0.08	<0.03	<0.002	<0.01	<0.006	<0.002	0.011 ± 0.003	0.047 ± 0.003	<0.02	<0.01	<0.02	0.016 ± 0.007	<0.003
RS-SS-4 NEAR SPENT FUEL AREA	<0.006	0.046 ± 0.005	0.03 ± 0.02	0.281 ± 0.007	<10	<0.02	<0.1	<0.04	<0.003	0.10 ± 0.02	0.04 ± 0.01	<0.003	0.452 ± 0.008	1.24 ± 0.01	<0.03	<0.01	<0.03	0.012 ± 0.009	<0.005
RS-SS-5 WEST OF TRANSFORMER YARD ROAD	<0.005	0.010 ± 0.004	<0.02	0.012 ± 0.003	<4	<0.02	<0.1	<0.03	<0.002	<0.01	<0.007	<0.002	0.022 ± 0.003	0.044 ± 0.003	<0.02	0.02 ± 0.01	<0.03	0.009 ± 0.007	<0.004
RS-SS-6 WEST END OF TRANSFORMER YARD	<0.002	0.010 ± 0.005	<0.03	0.024 ± 0.005	<6	<0.02	<0.1	<0.04	<0.002	<0.02	<0.008	<0.003	0.029 ± 0.004	0.070 ± 0.004	<0.03	0.05 ± 0.02	<0.03	<0.009	<0.005
RS-SS-7 SOUTH OF PERIMETER WALL 60 ft FROM REACTOR BUILDING	<0.005	0.045 ± 0.005	<0.02	0.104 ± 0.005	<6	<0.02	<0.1	<0.03	<0.002	<0.02	<0.007	<0.002	0.042 ± 0.004	0.086 ± 0.004	<0.03	<0.01	<0.03	<0.008	<0.004

TABLE C.7.5. Radionuclide Concentrations (pCi/g) in Rancho Seco Sediments - March, 1983

SEDIMENT	^{22}Na	^{46}Sc	^{54}Mn	^{58}Co	^{60}Co	^{51}Cr	^{65}Zn	^{103}Ru	^{106}Ru	^{108m}Ag	^{110m}Ag	^{124}Sb	^{125}Sb	^{126}Sn	^{134}Cs	^{137}Cs	^{144}Ce	^{152}Eu	^{154}Eu	^{155}Eu	^{166m}Ho
RS-SED-1 SOUTH RETENTION BASIN	<0.09	<3	155 ± 1	588 ± 1	785 ± 2	<200	<0.6	<4	8 ± 1	<0.06	43 ± 1	<4	69.5 ± 0.3	<0.06	67.8 ± 0.2	172 ± 1	<0.5	<0.2	<1	<0.1	<0.1
RS-SED-2 SOUTH RETENTION BASIN DISCHARGE PIPE	<0.1	215 ± 13	888 ± 3	4.290 ± 30	2.903 ± 5	<7.000	34 ± 6	<100	<14	<0.1	120 ± 10	<20	217 ± 3	<0.1	188 ± 2	341 ± 2	13 ± 7	<2	11 ± 8	<3	<0.1
RS-SED-3 SOUTH RETENTION BASIN	<0.007	0.03 ± 0.02	<0.006	<0.03	0.075 ± 0.006	<10	<0.02	<0.2	<0.04	<0.003	<0.02	<0.1	0.10 ± 0.01	<0.003	0.032 ± 0.006	0.72 ± 0.01	0.24 ± 0.04	0.06 ± 0.02	<0.03	0.04 ± 0.01	<0.005
RS-SED-4 SPRAY POND	<0.004	<0.02	<0.003	0.04 ± 0.02	<0.003	<1	<0.01	<0.1	<0.03	<0.002	0.02 ± 0.01	<0.06	<0.006	<0.002	<0.003	0.009 ± 0.004	<0.03	0.02 ± 0.01	<0.02	<0.007	<0.003

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TABLE C.7.6. Radionuclide Compositions Utilized for Inventory Construction

Radionuclide	Systems	Secondary Systems
^{54}Mn	0.69 μCi/cm^2	27 pCi/cm^2
^{55}Fe	4.8	710
^{57}Co	0.057	1.0
^{58}Co	4.0	110
^{60}Co	3.0	64
^{59}Ni	0.020	120
^{63}Ni	3.2	120
^{65}An	0.02	<0.4
^{90}Sr	<0.002	<0.009
^{106}Ru	<0.04	<4
^{108m}Ag	<0.002	<0.3
^{110m}Ag	0.78	<2
^{125}Sb	0.31	9.3
^{129}I	<2 x 10^{-6}	<0.0006
^{134}Cs	0.012	<0.3
^{137}Cs	0.057	0.4
^{141}Ce	<0.008	<4
^{144}Ce	<0.008	<0.6
^{238}Pu	6.5 x 10^{-5}	0.0023
^{239,240}Pu	3.3 x 10^{-5}	0.0012
^{241}Am	6.5 x 10^{-5}	0.0009
^{242}Cm	3.3 x 10^{-4}	0.0013
^{244}Cm	6.9 x 10^{-5}	0.0015

TABLE C.7.7. Estimated Surface Areas for Rancho Seco Reactor Systems

System	Surface Area (m^2)
Pressurizer	8.7×10^1
Piping (except RCS)	1.4×10^3
RCS Piping	1.9×10^2
Secondary System	2.3×10^5
Radwaste	1.4×10^3

TABLE C.7.8. Estimated Radionuclide Inventory Associated with Rancho Seco Operating Systems-March, 1983

(Inventory in curies (a,b)

Nuclide	Steam Generators	Pressurizer	RCS Piping	non-RCS Piping	Secondary	Radwaste
^{54}Mn	170	0.60	1.3	3.0×10^{-4}	0.062	9.7
^{55}Fe	1180	4.2	9.1	0.0078	1.6	67
^{57}Co	14	0.05	0.11	1.1×10^{-5}	0.0023	0.80
^{58}Co	990	3.5	7.6	0.0012	0.25	56
^{60}Co	740	2.6	5.7	7.0×10^{-4}	0.15	42
^{59}Ni	5	0.02	0.04	6.9×10^{-6}	0.0014	0.28
^{63}Ni	790	2.8	6.1	0.0013	0.28	45
^{65}Zn	4	0.02	0.038	$<4 \times 10^{-6}$	$<9 \times 10^{-4}$	0.28
^{90}Sr	<0.5	<0.002	<0.004	$<1 \times 10^{-7}$	$<2 \times 10^{-5}$	<0.03
^{106}Ru	<9	<0.03	<0.08	$<4 \times 10^{-5}$	<0.009	<0.6
^{108m}Ag	<0.5	<0.002	<0.004	$<3 \times 10^{-6}$	$<7 \times 10^{-4}$	<0.03
^{110m}Ag	1900	0.68	1.5	$<2 \times 10^{-5}$	<0.005	11
^{125}Sb	75	0.27	0.58	1.0×10^{-4}	0.021	4.3
^{129}I	<0.0005	$<2 \times 10^{-6}$	$<4 \times 10^{-6}$	$<7 \times 10^{-9}$	$<1 \times 10^{-6}$	$<3 \times 10^{-5}$
^{134}Cs	3	0.010	0.023	$<3 \times 10^{-6}$	$<7 \times 10^{-4}$	0.17
^{137}Cs	14	0.050	0.11	4.4×10^{6}	9.2×10^{4}	0.80

(a) Excluding neutron activated pressure vessel and internals, biological shield, concrete surfaces, residues, sludges and resins in tanks and spent fuel.

(b) Other long-lived radionuclides specifically listed in 10 CFR 61(13), e.g. ^{94}Nb, and ^{99}Tc were not included in the inventory because of their insignificant concentrations in the residual radioactive corrosion films in the plant piping and equipment.

TABLE C.7.8. (continued) Estimated Radionuclide Inventory Associated with Rancho Seco Operating Systems-March, 1983

(Inventory in curies (a,b)

Nuclide	Steam Generators	Pressurizer	RCS Piping	non-RCS Piping	Secondary	Radwaste
^{141}Ce	<2	<0.007	<0.02	$<7 \times 10^{-6}$	<0.001	<0.1
^{144}Ce	<2	<0.007	<0.02	$<7 \times 10^{-6}$	<0.001	<0.1
^{238}Pu	0.016	5.7×10^{-5}	1.2×10^{-4}	2.5×10^{-8}	5.3×10^{-6}	9.1×10^{-4}
$^{239-240}$Pu	0.008	2.9×10^{-5}	6.3×10^{-5}	1.3×10^{-8}	2.8×10^{-6}	4.6×10^{-4}
^{241}Am	0.016	5.7×10^{-5}	1.2×10^{-4}	9.9×10^{-9}	2.1×10^{-6}	9.1×10^{-4}
^{242}Cm	0.080	2.9×10^{-4}	6.3×10^{-4}	1.4×10^{-8}	3.0×10^{-6}	4.6×10^{-3}
^{244}Cm	0.017	6.0×10^{-5}	1.3×10^{-4}	1.7×10^{-8}	3.5×10^{-6}	9.7×10^{-4}
Total	4190	15	32	0.011	2.4	240

(a) Excluding neutron activated pressure vessel and internals, biological shield, concrete surfaces, residues, sludges and resins in tanks and spent fuel.

(b) Other long-lived radionuclides specifically listed in 10 CFR 61(13), e.g. ^{94}Nb, and ^{99}Tc were not included in the inventory because of their insignificant concentrations in the residual radioactive corrosion films in the plant piping and equipment.

TABLE C.7.9. Total Estimated Radionuclide Inventory at Rancho Seco Nuclear Station-March, 1983

Concentrations in curies (a,b)

Radionuclide	Half-life (years)	Inventory
^{54}Mn	0.855	180
^{55}Fe	2.7	1260
^{57}Co	0.739	15
^{58}Co	0.194	1060
^{60}Co	5.27	790
^{59}Ni	75,000	5
^{63}Ni	100	880
^{65}Zn	0.668	4
^{90}Sr	28.5	<0.5
^{106}Ru	1.01	9
^{108m}Ag	127	<0.5
^{110m}Ag	0.686	200
^{125}Sb	2.77	80
^{134}Cs	2.06	3
^{137}Cs	30.2	15
^{144}Ce	0.779	<2
^{238}Pu	87.7	0.016
^{239,240}Pu	24,100	0.008
^{241}Am	432	0.016
^{242}Cm	0.446	0.080
^{244}Cm	18.1	0.017
	Total	4460

(a) excluding neutron-activated pressure vessel and internals, biological shield, concrete surfaces, residues, sludges and resins in tanks, and spent fuel.

(b) Other long-lived radionuclides specifically listed in 10 CFR 61[13], e.g. ^{94}Nb, ^{99}Tc, and ^{129}I were not included in the inventory because of their insignificant concentrations in the residual radioactive corrosion films in the plant piping and equipment.

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APPENDIX D

APPENDIX D

SUMMARY OF PREVIOUS RESIDUAL RADIONUCLIDE INVENTORY ESTIMATES

D.1 REFERENCE BWR AND PWR POWER STATIONS D.1

D.1.1 Reference BWR D.1

D.1.2 Reference PWR D.2

D.2 RADIONUCLIDE INVENTORIES IN PREVIOUSLY DECOMMISSIONED REACTORS D.3

D.2.1 Ames Laboratory Research Reactor D.5

D.2.2 Elk River Reactor D.5

D.2.3 Fermi-1 .... D.5

D.2.4 North Carolina State University Research and Training Reactor D.5

REFERENCES D.6

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APPENDIX D

SUMMARY OF PREVIOUS RESIDUAL RADIONUCLIDE INVENTORY ESTIMATES

D.1. REFERENCE BWR AND PWR POWER STATIONS

Previous residual inventory assessments have been limited in number and have been extrapolated from a data base calculated from a theoretical standpoint, or from very limited data generated from the analysis of a single sludge sample. The two primary inventory estimates to date are published in NUREG/CR-130 (D1) and NUREG/CR-0672 (D2), which are decommissioning assessments for a reference PWR and a reference BWR, respectively. The PWR characterized was a single unit station of Westinghouse design, rated at 3500 MWt or 1175 MWe. The reference BWR was also a single unit station rated at 3320 MWt or 1155 MWe. The findings of these two assessments are summarized briefly by reactor type below.

D.1.1. Reference BWR

The total residual radionuclide inventory estimate associated with internal surfaces of piping and equipment at the reference BWR was 8500 curies, with 6300 curies associated with internal equipment surfaces and the remaining 2200 curies associated with internal piping surfaces. Of the 2200 curies present in piping, approximately 56% was calculated to be associated with the reactor water piping and 44% associated with the condensate piping. For the residual inventory of 6300 curies internally deposited in equipment, an estimated 30% was associated with equipment in the reactor building. The primary repositories were the reactor vessel corrosion film and the corrosion films in the residual heat removal heat exchangers. Nineteen percent of the radionuclide inventory internally deposited in equipment was in the turbine building, mainly associated with the condenser and the feedwater heaters. Approximately 51% of the internally deposited inventory was associated with equipment in the radwaste and control building, primarily with waste tanks.

The radionuclide inventories were calculated for equipment and piping in the reference BWR using a radionuclide composition which included the nuclides shown in Table D.1. The composition of the mixture was based upon a single BWR sludge sample which the authors indicate was similar to primary loop radwastes from other BWRs. Eighty-six percent of the estimated inventory at shutdown was due to two nuclides, ^{60}Co and ^{54}Mn. The ^{60}Co contributed almost half (47%) of the total radionuclide composition. Although the authors indicated ^{65}Zn can be an abundant radionuclide at BWRs, this nuclide constituted only 0.61% of their radionuclide composition.

The radionuclide composition also only took into account the gamma emitting nuclides contributing to the external gamma dose. Thus, beta, x-ray and alpha emitting radionuclides were not included in the inventory estimates. This excluded potentially significant long-lived contributors to the overall inventory such as ^{55}Fe, ^{63}Ni, ^{59}Ni, ^{90}Sr and the transuranics.

TABLE D.1. Radionuclide Composition Utilized for Estimating Reference BWR and Reference PWR Piping and Equipment Inventories. (Relative Abundances in percent of total at Shutdown)

Radionuclide	BWR Composition	PWR Composition
^{51}Cr	2.1	26.9
^{54}Mn	39.	4.2
^{59}Fe	2.5	1.7
^{58}Co	0.93	50.1
^{60}Co	47.	8.2
^{65}Zn	0.61	
^{95}Zr	0.41	
^{95}Nb	0.40	
^{103}Ru	0.23	
^{106}Ru	0.28	
^{134}Cs	1.9	
^{137}Cs	3.4	
^{141}Ce	0.30	
^{144}Ce	0.81	

D.1.2. Reference PWR

The total radionuclide inventory associated with the interior of the reference PWR reactor system was estimated to be 4800 curies. The steam generators were estimated to contain approximately 90% of the total deposited inventory, or 4400 curies. The corrosion films associated with the reactor coolant system piping and the reactor vessel internal surfaces each constituted 3% of the total surface-deposited inventory. The piping other than the reactor coolant system contained slightly over 1% of the total estimated inventory. The radionuclide composition at shutdown utilized for the reference PWR inventory estimates is shown in Table D.1 and, as shown, was very limited in scope. Only three of the five nuclides listed in the table were actually utilized in the inventory construction, ^{51}Cr and ^{59}Fe being excluded from consideration because of their short half-lives. Because this inventory was estimated based upon external gamma dose rates, a substantial number of potentially significant radionuclides were not included in the reference PWR inventory. These include ^{59}Ni and ^{63}Ni which, because of the extensive use of inconel (80% Ni) in PWR primary systems, will be relatively abundant. Also excluded were ^{55}Fe, ^{90}Sr, and the transuranics. An additional shortcoming for the reference PWR inventory is that it contains no estimate for the radioactive residue which would be contained in the chemical and volume control system. It might be expected that a significant fraction of the inventory, at least 10%, should be associated with this system, especially since the radwaste system in the reference BWR constituted such a large portion of the total inventory.

D.2. RADIONUCLIDE INVENTORIES IN PREVIOUSLY DECOMMISSIONED REACTORS

Although no large commercial nuclear electric generation facilities have been removed from service and dismantled to date, substantial decommissioning experience has been gained at several smaller facilities including early demonstration units such as Elk River (D3) and Pathfinder (D4), as well as research reactors such as the Ames Laboratory Research Reactor (D5) and the North Carolina State University Research and Training Reactor (D6). The Enrico Fermi liquid metal demonstration unit has also been decommissioned and placed into a safe storage condition (D7).

The residual radionuclide inventories associated with the decommissioning of four facilities are shown in Table D-2. The composition shown is in percent of the total inventory and the total inventory in curies for the radwaste generated during the decommissioning is also shown in the table.

TABLE D.1. Radionuclide Composition Total Inventory Associated with Radwastes Generated During Decommissioning of Four Reactors. (Compositions in percent of the total and total inventory in curies.)

Radionuclide	Elk River	Ames Lab	N.C. State	Fermi-1
^{3}H	-	1.1	-	-
^{54}Mn	0.34	-	-	-
^{55}Fe	55	53	0.01	68
^{60}Co	42	40	6.1	29
^{63}Ni	2.3	4	30	1.8
^{65}Zn	-	0.9	-	-
^{108}Ag	0.6	-	-	-
^{110m}Ag	-	0.5	-	-
^{109}Cd	-	0.2	0.3	-
^{133}Ba	-	-	2	-
^{152}Eu	-	0	62	-
Total Inventory in curies	10,000*	6,900*	2.3*	5,200**

*Includes reactor pressure vessel and internals
**Waste from only a partial decommissioning

The North Carolina State University Research and Training Reactor was anomalous in comparison to the other three reactors presented in Table D.2. The total inventory was unusually low (2.3 curies), and the radionuclide composition was dominated by ^{152}Eu which represented over 60% of the total inventory.

The other three reactors shown had similar residual radionuclide compositions in the radwaste generated during decommissioning. Iron-55 was the most abundant radionuclide in all three, ranging from 53 to 68 percent of the total and averaging 59%. The second radionuclide in abundance in these three reactors was ^{60}Co, which ranged from 29 to 42% of the total and averaged 37%. Nickel-63 was the third most abundant radionuclide, which ranged from 1.8 to 4% of the total inventory and averaged approximately 3%. These three radionuclides, ^{55}Fe, ^{60}Co, and ^{63}Ni, constituted over 95% of the total reported in the Elk River, Ames Laboratory, and Fermi-1 radwastes generated during reactor decommissioning.

A comparison of inventory versus unit rating and years of operation is shown in Table D.3. The total inventories associated with these four reactors do not scale well with the product of years of operation and the unit size on a MWt basis. However, this is not surprising since these were experimental reactors of highly variable design and materials of construction which would greatly affect the radionuclide composition and amounts produced. The four units ranged from 10 kW, in the case of the N.C. State University Research and Training Reactor, to 430 MWt in the case of Fermi-1.

TABLE D.2. Comparison of Total Inventory Associated with Radwaste Generated During Decommissioning with Unit Size for Four Reactors

Reactor Facility	Unit Size (MWt)	Years of Operation	Total Inventory of Decommissioning Waste (curies)	Curies/MW·yr
N.C. State Univ.	0.010	13	2.3*	18
Ames Laboratory	5	10.5	6,900*	131
Elk	58	5.2	10,000*	33
Fermi-1	430	9	5,200**	---

*Includes pressure vessel and internals
**Represents waste from only a partial decommissioning

Fermi-1 was only partially decommissioned, it was only placed into long term safe storage, therefore the inventory generated and presented in Table D.3 obviously does not represent the total radionuclide inventory associated with this facility. The four reactors are described briefly below along with the decommissioning activities.

D.2.1. Ames Laboratory Research Reactor

The Ames Lab reactor was a heavy water moderated facility which utilized 93% enriched uranium. It operated in the full power research mode from June, 1966 until the end of December, 1977. Decommissioning was initiated immediately after shutdown and is described in several documents (IS-4789) (D8), NUREG/CR-3336 (D5), and Link and Voigt, 1982) (D9). Total wastes generated during the decommissioning amounted to 6900 curies, of which over 5500 curies was associated with the thermal shield. Primary activities associated with the shield were ^{60}Co, ^{55}Fe and ^{63}Ni.

D.2.2. Elk River Reactor

The Elk River Reactor Power Station was completed in 1960 as a part of the AEC reactor demonstration program. The reactor was a BWR of Allis Chalmers design, rated at 58 MWt. Initial criticality was in November, 1962, and final shutdown was at the end of January, 1968. Decommissioning was conducted in three phases: 1) Planning, 2) Dismantlement, and 3) Facility Closeout. The overall decommissioning cost was approximately $6 million in 1972. Decommissioning entailed total dismantlement, with the site being returned to conditions as existed prior to reactor installation. An estimated 10,000 curies were disposed of during the decommissioning activities. Included in the total were the reactor vessel and internals. The decommissioning activities are described in two reports, COO-651-93 (D10) and NUREG/CR-2985) (D3).

D.2.3. Fermi-1

The Fermi-1 reactor was an experimental demonstration liquid metal fast breeder which started operations in 1963 and terminated operations in 1972. Decommissioning of the facility entailed placement into safe storage, during which approximately 5000 curies of radwaste were disposed of. Total decommissioning costs incurred were slightly over $7 million in 1974 dollars.

D.2.4. North Carolina State University Research and Training Reactor

This reactor operated from March, 1950, until mid-February, 1973, and was rated at 10 kWt. The decommissioning entailed total dismantlement and removal of the reactor facility. The radionuclide content associated with the waste generated during decommissioning was minimal due to the small size of the reactor. The decommissioning activities, schedule and costs are contained in NUREG/CR-3370 (D6) and other technical reports available from North Carolina State University.

References

D1. Smith, R. I., G. J. Konzek and W. E. Kennedy, Jr. Technology Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station. NUREG/CR-0130, Vol. 1 and 2. Prepared for the U. S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, 1978.

D2. Oak, H. D., G. M. Holter, W. E. Kennedy, Jr., and G. J. Konzek. Technology, Safety and Costs of Deccmmission a Reference Boiling Water Reactor Power Station. NUREG/CR-0672, Vol. 1 and 2. Prepared for the U. S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington, 1980.

D3. Miller, R. L. and J. A. Adams. Evaluation of Nuclear Facility Decommissioning Projects. Project Summary Report. Elk River Reactor, NUREG/CR-2985. U. S. Nuclear Regulatory Commission, Washington, D.C., 1982.

D4. Bjeldanes, M.N. Pathfinder Conversion. Transactions of the American Nuclear Society. 12 (Supl.): 52-53, 1969.

D5. Link, B. W. and R. L. Miller. Evaluation of Nuclear Facility Decommissioning Projects. Summary Report. Ames Laboratory Research Reactor, NUREG/CR-3336, U.S. Nuclear Regulatory Commission, Washington, D.C., 1983.

D6. Link, B. W. and R. L. Miller. Evaluation of Nuclear Facility Decommissioning Projects. Summary Report. North Carolina State University Research and Training Reactor, NUREG/CR-3370, U. S. Nuclear Regulatory Commission, Washington, D.C., 1983.

D7. Miller, R. L. and B. W. Link. Evaluation of Nuclear Facility Decommissioning Projects. Summary Report. Enrico Fermi-1 Reactor, NUREG/CR-3116, U. S. Nuclear Regulatory Commission, Washington, D.C., 1983.

D8. Reactor Staff, Final Report, Decommissioning of the Ames Laboratory Research Reactor, IS-4789, UC-78a, January, 1982. Available from NTIS, Springfield, VA.

D9. B. W. Link and A. F. Voigt. Decommissioning of Ames Laboratory Research Reactor. Proceedings of the 1982 International Decommissioning Symposium, Seattle, WA, October 10-14.

D10. United Power Association. Final Elk River Reactor Program Report. COO-651-93, November, 1974. Contract AT (11-1)-651, NTIS, Springfield, VA.

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The residual radionuclide concentrations, distributions and inventories at seven nuclear power plants (four shutdown and three operating) have been investigated to provide a data base for use in formulating policies, strategies and guidelines for the eventual decommissioning of retired nuclear power plants. This study has addressed radionuclides (both activation and fission products) transported from the reactor pressure vessel and deposited in all other contaminated systems of each nuclear plant.

Emphasis has been placed on measuring the long-lived radionuclides which are of special concern from a low-level waste management standpoint, including ^{60}Co, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{94}Nb, ^{99}Tc, ^{129}I, ^{137}Cs, and alpha-emitting transuranic radionuclides with half-lives greater than five years.

The total residual radionuclide inventories (excluding the pressure vessel) at the seven nuclear plants examined in this study appear to be proportional to the product of the unit power level (megawattage) and the length of operations in years. Thus, extrapolations of the radionuclide inventories at other nuclear plants may perhaps be made. For the most part, the radionuclide compositions and inventories measured in this program were in reasonably good agreement with the limited data base used in the earlier conceptual assessment studies of the technology, safety, and costs of decommissioning a reference PWR (NUREG/CR-0130) and a reference BWR (NUREG/CR-0672). Thus, the conclusions reached in these conceptual studies from a radiological standpoint will essentially remain unchanged.

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